

1.0 Project Mission and Requirements

1.1 Project Mission

The mission of the Sandia National Laboratories, (SNL) Radiation Protection Internal Dosimetry (RPID) Project is to detect, assess, document, and provide appropriate response to occupational exposures from internally deposited radioactive materials originating from SNL activities. This technical basis document (TBD) is designed to meet these objectives by providing guidance on the following:

- Define the project participation requirements,
- Describe the use of defensible bioassay methods and frequencies,
- Determine appropriate bioassay analytical techniques and associated quality control/assurance measures,
- Describe a defensible approach to determine the internal dosimetry of detected exposures,
- Develop reporting levels and define appropriate responses to occurrences,
- Describe the documentation control and quality control/assurance requirements, and
- Provide a technical basis for the generation of Procedures.

This project will meet, or exceed, the requirements specified in the Department of Energy (DOE) Radiological Control Manual (DOE/EH-0256C) and promulgated under Title 10 of the Code of Federal Regulations Part 835 (10 CFR 835). The SNL Radiological Control Manual (RCM) administers the DOE Radiological Control Manual(DOE/EH-0256C) at SNL.

1.2 Project Scope

The RPID Project will be implemented at all SNL facilities for activities involving the processing and/or storing of radioactive materials. This project includes activities at the Tech Area (TA) I, TA II, TA III, TA IV, TA V, Coyote Test Field, and environmental restoration sites at SNL, located in Albuquerque, New Mexico, and the Kauai Test Facility(SNL/KTF). Reference to SNL throughout this document includes facilities and activities at the Albuquerque location and at SNL/KTF.

1.3 Project Implementation Strategy

Activities at SNL facilities involve numerous processes handling radioactive materials which may result in occupation internal exposures. This document provides the technical basis for implementing the RPID Project which is designed to monitor these exposures. The TBD will specifically address internal dosimetry aspects of common contaminants of potential concern (COPC) at SNL (e.g., tritium, uranium and fission/activation products). Additional Internal Dosimetry requirements for facilities and/ or special operations may be specified in radiation work permits(see procedure RPOP-06-605).

1.4 Project Responsibilities

The manager of the Radiation Protection Measurements Department (7715), or his designee, is responsible for all aspects of the RPID Project. All internal dosimetry records are kept by the RPID Project and are administered by the radiation Protection Records Information (RPRI) Project. Arrangements for internal monitoring and subsequent reports are also the responsibility of Department 7715. The Occupational Medical Center(3300) is responsible for occupational health issues and will be kept informed on relevant findings of the RPID Project. In addition, all SNL radiological workers are responsible for complying with the procedural guidance given in the Radiation Protection Procedure Manual(RPPM).

1.5 Project Requirements

10 CFR 835 and the DOE RCM provide specifications and requirements for internal dosimetry projects developed for DOE facilities. 10 CFR 835 states occupational annual dose limits, requires dose assessment, and defines participation, reporting, recording, and quality control requirements. Reference to radiological workers in this TBD includes SNL employees and the employees of companies under contract to SNL. The RCM introduces administrative control levels below 10 CFR 835 limits, lifetime dose control levels, and special control levels for individualized exposure control. Mandatory requirements in 10 CFR 835 are enforceable by law and are identified through the use of shall. Requirements contained in the RCM are mandatory to the extent they are incorporated by a contract or through administrative means.

1.5.1 Dose Limits and Assessment Requirements

10 CFR 835 requires that the occupational exposure of radiological workers to radiation and radioactive materials from routine DOE activities shall not exceed the following:

- The annual limit for stochastic effects, measured as the total effective dose equivalent (TEDE) from both internal and external sources, is 5 rems (0.05 Sieverts),
- The annual dose equivalent limits for non-stochastic effects are 15 rems (0.15 Sieverts) to the lens of the eye, 50 rems (0.5 Sieverts) shallow dose to the extremities or skin, and 50 rems (0.5 Sieverts) to any organ or tissue.

An annual occupational exposure limit of 5 Rem(0.05 Sievert) is specified in 10CFR835.202. The RCM further limits worker exposure to 2 Rem(0.02Sievert) annually. At SNL the RPPM describes a series of Administrative Control Levels(ACLs) which restrict most workers to 0.1 Rem (0.001 Sievert) annually. Management approval is required to exceed this limit. Higher exposure limits require higher levels of management approval. This approval requires consideration of alternative measures and justification for the higher levels. Allowable non-stochastic dose limits in the RP Program are identical to the 10 CFR 835 requirements.

The RCM further defines a lifetime control level (LCL) of N rem (0.01 N Sieverts) where N is the age of the person in years. Special control levels shall be established for personnel who have

lifetime doses greater than N rem. The goal of the RPID Project is for all employees to comply with the LCL requirement. A dose reconstruction project may be required to evaluate the LCL for some workers at SNL. Such a project will be addressed in another TBD.

10 CFR 835 requires that the following limits shall be observed in these situations:

- The annual dose limit for the embryo/fetus of a declared pregnant radiological workers is 500 mrem (0.05 Sieverts) from conception to birth,
- The annual TEDE for minors and students under the age of 18 is 100 mrem (0.01 Sieverts), and
- The annual TEDE for members of the public entering controlled areas is 100 mrem (0.01 Sieverts).

The RCM provides an additional recommendation that dose rates be maintained less than 50 mrem (0.0005 Sievert) per month to declared pregnant radiological workers. The RPID Project adopts all of the 10 CFR 835 and RCM pregnant radiological worker and visitor dose limits.

10 CFR 835 states that internal doses shall be assessed and reported, and that internal dosimetry projects shall be capable of demonstrating compliance to the 10 CFR 835 annual dose limits. The TEDE shall be determined by summing the committed effective dose equivalent (CEDE) from internally deposited radionuclides and the effective dose equivalent (i.e., the deep dose equivalent) from external sources of radiation.

10 CFR 835 contains both primary and secondary limits for occupational exposures to external and internal sources of ionizing radiation in the workplace. The primary limits for internal exposures were previously stated and are expressed in terms of CEDE. Secondary standards are derived from these limits and include annual limits of intake (ALI) and derived air concentrations (DAC). These values were chosen to provide safe working conditions and to maintain exposures within the primary standards. Compliance of the primary 10 CFR 835 standards shall be demonstrated with bioassay data rather than air monitoring data unless the following are true:

- Bioassay data are unavailable,
- Bioassay data are inadequate, or
- Internal dose estimates based on representative air concentration values are demonstrated to be as or more accurate.

The RPID Project is designed to meet the 10 CFR 835 and RCM dose assessment requirements.

1.5.2 Dose Monitoring Requirements

10 CFR 835 specifies that the monitoring of internal radiation exposures shall be performed on the following individuals:

- Radiological workers who, under typical conditions, are likely to receive 100 mrem (0.001 Sievert) CEDE or greater from all occupational intakes in the year,
- Declared pregnant workers likely to receive an intake resulting in a dose equivalent to the embryo/fetus exceeding 50 mrem (0.0005 Sieverts),
- and, Minors (i.e., individual less than 18 years of age) or members of the general public who are likely to exceed annual CEDE exceeding 50 mrem (0.0005 Sieverts).

The ability of the internal dosimetry project to comply to 10 CFR 835 and RCM requirements may be effected by the presence of radioactive materials from previous intakes. Efforts shall be made to obtain records of any prior exposures. Baseline bioassay measurements should be performed on individuals with previously documented exposures. In addition, baseline measurements are recommended for all SNL radiological workers.

The RCM specifies these additional requirements for the following situations:

- Baseline bioassay monitoring of individuals who are likely to receive intakes greater than 100 mrem CEDE shall be conducted prior to potential occupational exposures,
- Bioassay should be performed when air sampling results indicates potential exposures exceeding 100 mrem,
- Bioassay should be performed whenever facial and nasal contamination is detected (i.e., indicating potential internal contamination),
- Bioassay should be performed whenever internal exposure is suspected by the Radiological Control Organization,
- and, Termination bioassay monitoring should be performed on bioassay project participants prior to employment termination.

The RPID Project shall implement both the 10 CFR 835 monitoring requirements and the RCM technical guidance procedures.

Bioassay frequencies should be sufficient to ensure that any dose limits are not being exceeded. The RPID Project will adopt bioassay frequencies which are adequate to detect exposures of 100 mrem (0.001 Sieverts) CEDE whenever technically and practically feasible.

1.5.3 Dose Management Requirements

In situations where the worker's exposure exceeds or is close to exceeding the ACLs, future exposure shall be carefully controlled. In these cases additional monitoring requirements may be required to control internal exposure.

1.5.4 Records Requirements

10 CFR 835 states that all individual internal and external dose evaluations shall be recorded. This stipulation includes all personnel dosimetry assessments performed, including evaluations

not explicitly required by 10 CFR 835. The records shall be sufficient to evaluate compliance and provide information for dose reports. The dosimetry records shall include the following quantities for internal dose resulting from intakes received during the year:

- CEDE,
- Committed dose equivalent to any organ, or tissue, of concern,
- and, Estimated intake and identity of radionuclides for the evaluated work year.

Records shall include the following quantities for the summation of internal and external dose:

- Total annual effective dose equivalent,
- The sum of the dose equivalent from external radiation exposures with the committed dose equivalent for any organ, or tissue, assigned an internal dose during the year, and
- The cumulative TEDE received while employed at the site or facility since January 1, 1989. Written estimates which are agreed upon by the monitored individual may be used in the absence of formal records of previous occupational exposures.

Records shall also include total doses from internal and external sources to the embryo/fetus of declared pregnant radiological workers. Records shall be kept of the TEDE received by minors, students, and members of the general public entering a controlled area.

Data necessary to allow verification, correction, or recalculation of recorded doses (e.g., survey results, measurements, calculations, etc.) shall be generated and recorded. Records required by 10 CFR 835 shall be available to the monitored individual upon request and shall be transferred to the DOE upon cessation of activities at the site that could cause exposures. All records shall be retained until final disposition is authorized by the DOE unless otherwise specified in 10 CFR 835 Subpart H.

The RPID Project will adopt all of the record requirements promulgated under 10 CFR 835. Exposure records and individual health hazard case files are compiled and retained by the Radiation Protection External Dosimetry(RPXD) Project.

1.5.5 Reporting Requirements

10 CFR 835 requires that each individual monitored for internal dose shall be provided a radiation dose report annually. The report shall include the following:

- CEDE for the year,
- Committed dose equivalent to any organ, or tissue, of concern,
- and, Estimated intake and identity of radionuclides.

Situations in which there was no detectable internal doses or intakes of radionuclides will be reported as such.

In addition, terminating employees shall be provided records of their dose within 90 days of termination of employment. The termination report should include the total effective dose for the year, the cumulative TEDE, and the lifetime occupational dose. A written estimate shall be provided at the time of termination if requested. If an internal dose evaluation is still in progress at 90-day limit, the worker should be notified and provided with an interim report.

Reports are controlled and generated by the RPXD Project in accordance with the requirements specified in 10CFR835.

1.5.6 Quality Assurance Requirements

The internal dosimetry project shall be adequate to demonstrate compliance with the requirements promulgated under 10 CFR 835. Therefore, internal audits of all functional project elements shall be conducted. The frequency of project audits shall not exceed 3 years and shall include project content and implementation. Quality assurance practices will be designed to identify project deficiencies and initiate corrective actions. Audits should be performed by both internal and independent, external reviewers. In addition to the 10 CFR 835 requirements, the RCM includes auditing independent contractor internal dosimetry projects whenever applicable. Auditing frequencies will be, at a minimum, every 3 years.

The RPID Project will contain quality assurance provisions including a project re-evaluation frequency, at a minimum, of three years.

Additional Quality Assurance is provided for the elements of the RPID Project with the greatest potential to impact exposure assessments. This includes software used to calculate exposures, lab processes used during Radiobioassay, and the operation of the whole body counting system.

Internal dose calculations are performed using the software INDOSE. This software is commercially available and is validated and verified by the supplier. The INDOSE manuals provide an example hand calculation along with results from previous INDOSE runs. Software quality assurance is provided by confirmation runs on RPID project computers using the input given in the manual and which yield the expected results. Further assurance of calculation repeatability is provided by the printed output which displays all user controlled parameters used in the calculation. A review of the calculational inputs is sufficient to confirm the results.

Radiobioassay for excreta is performed either onsite at SNL or off-site by a contract facility. In both cases the following elements are in place providing quality assurance of the sample counting results. These same elements are also applicable for in-vivo bioassay(whole body counting).

- Operating procedures,
- Staff training requirements,

- Quality control requirements for counting systems,
- Performance requirements(required MDAs),and
- A Quality Assurance Program.

For SNL facilities, both radiochemistry and whole body counting, these elements are *required by* Contractor laboratories are required to have these program elements in place in order to be awarded a contract.

2.0 Principles and Selection of Bioassay Techniques

Internal dosimetry is the determination of the kind, quantity, location, and retention of radionuclides in the body (i.e., bioassay), and the assessment of dose from these exposures. Internal dosimetry programs serve a multitude of purposes including:

- Exposure Control (i.e., detection of intakes),
- Dose Assessment from Intakes,
- Documentation of Results,
- and, Regulatory Compliance Demonstration.

Workplace radiological controls should be designed to minimize, or prevent, occupational exposures to radionuclides to the extent reasonably achievable. However, internal exposures may occur when these measures are not adequate, or not followed. The SNL/NM Internal Dosimetry Program is designed to detect exposures whenever radioactive material containment is compromised. Therefore, The SNL/NM Internal Dosimetry Program serves as a final quality control check of the various radiation protection programs in place at SNL/NM (e.g., Air Sampling Program, Sealed-Source Evaluation Program, etc.).

The optimal internal dosimetry program is process specific and is expected to differ among the various SNL/NM facilities. This Section provides programmatic guidance for the proper selection of bioassay techniques for the SNL/NM Internal Dosimetry Program.

2.1 Selection of Bioassay Techniques

Selecting the proper bioassay technique depends on the potential exposure routes, and the physical and chemical nature of the COPC at each facility. Figure 2-1 illustrates the primary routes of internal exposure and the subsequent biokinetic fate (e.g., absorption, elimination, deposition, etc.) of the radioactive material (NCRP 87).

2.1.1 Exposure Pathways

The inhalation pathway is expected the most significant exposure route at SNL/NM. Prevention of inhalation exposure requires the use of respiratory protection devices once airborne radioactive materials are detected. However, detection is difficult, especially for low-level, chronic releases, and is often accomplished retrospectively (i.e., after the release event had occurred). The initial deposition is dependent on the physical form (e.g., particle size) of the inhaled COPC, and the anatomic and physiologic status of the exposed individual. The subsequent metabolism of deposited materials is a function of the deposition site and its physical and chemical properties. For example, large particles of soluble radioactive materials are deposited in the upper respiratory tract where they can be directly absorbed or rapidly cleared by mucociliary activity into the gastrointestinal (GI) tract. In contrast, small

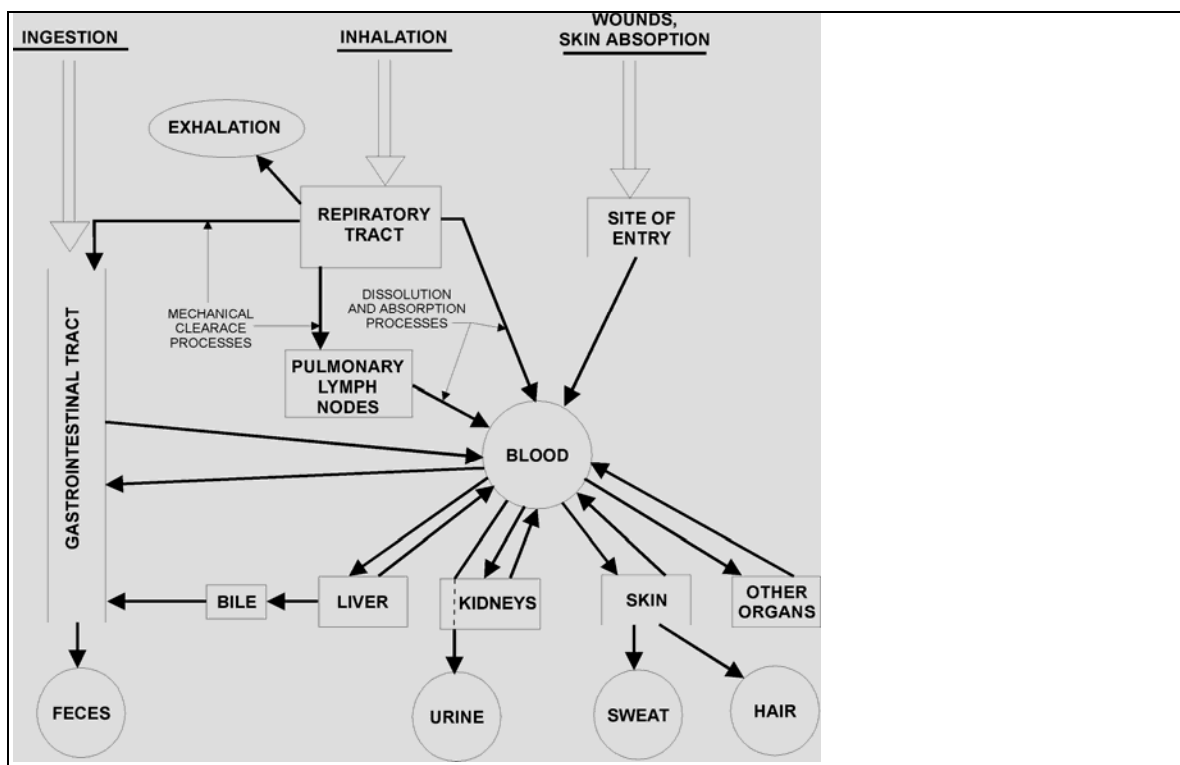


Figure 2-1 Exposure Pathways and Metabolic Fates

particles of insoluble materials are deposited in the pulmonary region of the lung where they may remain for long time periods.

Internal exposures from the ingestion pathway can be minimized through proper work practices such as personal protective gear (e.g., gloves) and proper occupational hygiene techniques (i.e., hand-washing, not-eating food in controlled areas, etc.). The magnitude of ingestion exposures are dependent on the solubility of the consumed COPC. Highly soluble materials (e.g., tritiated water, radioactive iodides, etc.) can be rapidly absorbed by the GI system into the body where it may be deposited in specific organs. However, insoluble COPC forms may be minimally absorbed and pass through the GI system, ultimately eliminated as feces.

The skin-absorption pathway can also be minimized through proper work practices. Although the skin provides an effective barrier against many insoluble COPC, highly soluble materials (e.g., radioactive organic solvents, tritiated water, etc.) may be easily absorbed resulting in a potentially significant source of internal exposures.

Wound exposures result from accident situations which are rare at properly designed facilities. However, this pathway may result in high internal exposures from insoluble materials since the skin is circumvented. Wound exposures have resulted in the highest internal doses at most DOE sites. Once deposited in the wound, the subsequent metabolism of the COPC are dependent on its solubility properties.

2.1.2 Biokinetic Pathways

Once introduced into the body, the radioactive material is transferred throughout the body by systemic circulation (i.e., blood) where it may be retained by various organs or eliminated by excretion, perspiration, exhalation, secretion, or exfoliation. Radionuclide materials can then be metabolized into other compounds which may result in different deposition and elimination patterns. Bioassay procedures are designed to capitalize on the characteristic biokinetic pathways of the various COPC. The appropriate bioassay can be determined by considering the following:

- Ease of detection of internally deposited materials,
- Distribution and elimination of the material within the body,
- Temporal changes in deposition and elimination,
- Required analytical sensitivity,
- and, Reliability of biokinetic models.

Bioassay techniques are generally classified as *in-vivo* and *in-vitro* techniques. *In-vivo* measurement involves the external measurement of radiation emissions from internally deposited COPC. This technique is the most direct method for estimating COPC body burdens. However, not all radionuclide COPC emit radiations which can be externally detected. Since direct analysis of tissue samples is seldom possible in living persons, bioassay must be evaluated using *in-vitro* techniques. *In-vitro* techniques involve the collection and assay of samples excreted, secreted, or removed from the body. Encircled objects in Figure 2-1 represent potential *in-vitro* bioassay samples. Both *in-vivo* and *in-vitro* bioassays possess unique advantages and deficiencies. A combination of *in-vivo* and *in-vitro* bioassay techniques will be used in the SNL/NM Internal Dosimetry Program.

2.2 In-Vivo Bioassay Techniques

In-vivo measurements are possible for radionuclide COPC which emit photons with sufficient energies to be detected outside of the body (e.g., radium-226, cobalt-60, etc.). These assays provide direct measurements of the subject's body burden, eliminating any dependency on biokinetic models and their associated uncertainties. However, the sensitivity and accuracy of *in-vivo* techniques may be limited by the following:

- Background radionuclide interference within the subject (e.g., potassium-40, cesium-137, etc.),
- Variability of photon attenuation and radionuclide deposition patterns between the measured worker and the calibration geometries,
- Potential errors caused by the presence of skin contamination on the subject,
- and, Background interferences from environmental sources (e.g., cosmic rays, radon progeny, etc.).

These limitations are less significant for higher energy emitting radionuclides (e.g., cobalt-60, iodine-131, etc.) and radionuclide compounds deposited near the body surface. *In-vivo* assay may also be useful in monitoring insoluble forms of low-energy photon emitters deposited in the lungs (e.g., oxides of transuranics, etc.) which are often difficult to measure using *in-vitro* techniques.

In-vivo techniques are primarily used for routine surveillance of workers to detect unknown exposures. Whole-body, and possibly organ-specific, body burdens can also be evaluated when the counting equipment is properly calibrated. The corresponding dose and retention patterns can also be determined in these situations. *In-vivo* counting systems are generally classified as whole-body counters (WBC) and organ-specific counters.

2.2.1 Whole-Body Counters

Several counting systems are available for WBC, ranging from a single, unshielded detector to multi-detector arrays used in shielded laboratories designed to minimize the influence of environmental background radionuclides. Detectors available for *in-vivo* measurements include inorganic crystals (e.g., sodium iodide (NaI), phosphor, etc.), solid state (e.g., high purity germanium (HPGe),), gas-flow proportional, and organic scintillants. Selection of the appropriate equipment is dependent on the required sensitivity, number of radionuclides potentially present, and the photon emission characteristics.

The SNL/NM Radiation Protection Measurement Department (7715) maintains an *in-vivo* bioassay capability using a HPGe shadow-shield detection system. This system is capable of reliably quantifying body burdens of high energy photon emitting radionuclides (i.e., photon emissions greater than 250 KeV). In addition, the Department 7715 system provides a qualitative spatial distribution of internal depositions. The use of this system is emphasized in the SNL/NM Internal Dosimetry Program.

2.2.2 Organ-Specific Counters

Organ counters can be used to detect and quantify radionuclide depositions in specific organs. Examples of these systems are lung counters, thyroid probes, and wound counters. Lung counters are design to determine body burdens of insoluble particles deposited within the lungs. Such depositions are slowly eliminated from the body and may be difficult to detect using *in-vitro* bioassay methods. The specialized facilities required for determining lung burdens are not currently available at SNL/NM. Suspected lung exposures will be assessed by an outside laboratory (i.e., Los Alamos National Laboratories).

Radionuclide compounds containing iodine may be selective collected and retained in subject's thyroid gland. Thyroid burdens are determined by placing a radiation detector close to the subject's neck. The neck measurements can then be compared to the results from a standard thyroid phantom containing the radionuclide of interest (e.g., iodine-131, iodine-125, etc.). Department 7715 maintains a HPGe probe which is capable of qualitative assessments of thyroid

burdens. A nearby nuclear medicine facility (e.g., Albuquerque Veterans Administration Medical Center) may be used when quantification is desired.

Bone seeking radionuclides can be detected, and sometimes quantified, by measuring limb or skull activities. Skull counting may be particularly useful because of the lack of attenuating tissue, the large radius of curvature large allows close approach of detector, and the large skeletal mass associated with skull. However, calibration of these detection systems is difficult, and measurements require specialized facilities which are not available at SNL/NM. The use of bone counting is limited in the SNL/NM Internal Dosimetry Program to accident situations and requires the use of outside laboratories.

Another important use of organ-specific counters is the assessment of wound exposures. Depositions of soluble materials would be quickly transported from the wound site and may be quantified using *in-vitro* techniques. However, insoluble depositions may remain at the wound site and can be assessed using an appropriate probe. The information obtained is useful to evaluate the need for medical intervention (e.g., tissue excision) since insoluble materials will be eventually absorbed if permitted to remain. Department 7715 HPGe probe which will be used to assess wound depositions in the SNL/NM Internal Dosimetry Program.

2.3 In-Vitro Bioassay Techniques

In-vitro techniques indirectly determine the radionuclide body burden by measuring biologic samples outside of the body. *In-vitro* techniques are performed under controlled conditions (e.g., low radiation backgrounds, constant measurement geometries, etc.) which result in lower analytical sensitivities compared to *in-vivo* techniques. In addition, radionuclides which do not emit penetrating photons can be assessed. However, the accuracy of *in-vitro* assessments is limited by the uncertainties associated with the biokinetic model used. Potential uncertainties include:

- Extrapolation of biokinetic data from animals to humans,
- Variability among individuals (i.e., deviances from "Standard Man" assumptions),
- Effects of the individual's health on biokinetic patterns,
- Establishing the correct chemical and physical form of the radionuclide compound,
- Establishing time of exposure and type of exposure (i.e., chronic versus acute),
- Using models based on data from assumed similar compounds,
- and, Analytical measurement uncertainties.

Biokinetic models are typically expressed in the form of retention functions. Retention functions range from simple exponential descriptions of compounds distributed uniformly throughout the body to multiple exponential functions, power functions, and their combinations. Differentiating these retention functions yields the biologic rate of excretion which can be used to estimate body burdens from *in-vitro* analytical results.

2.3.1 Urine Analysis

A fraction of the radionuclide body burden absorbed into systemic circulation is removed by the kidneys and discharged from the body as urine. This removal fraction is dependent on the compound's chemical form (i.e., ionic or complexed form) and is subject to temporal changes as original deposition locations release materials by kinetic processes (e.g., cellular turnover, compound translocation or metabolism, etc.) into the systemic circulation. The magnitude of the initial body burden and its subsequent retention in the body can be determined indirectly from urine analysis (a.k.a. urinalysis) results.

Urinalysis serves several purposes in internal dosimetry programs including the following:

- Detect internal exposures not detected by other means (e.g., air sampling, *in-vivo* bioassay, etc.),
- Provide repetitive estimates of the body burdens accumulated in the working population,
- Assess the performance of radiation protection control practices,
- and, Provide the basis for estimating transferable body burdens or refining biokinetic models.

Urinalysis is the most common *in-vitro* bioassay technique since samples can be easily collected and their subsequent analysis can be extremely sensitive. Considerable variability may be observed when using general biokinetic models to evaluate urine samples from a single individual or from different individuals. Physiological uncertainties can be introduced using single voiding samples which reflect the biologic conditions at the time of sampling and may not be representative of the long-term conditions within the body. Other sources of uncertainty include time of exposure, multiple exposures, and the effects of medical intervention techniques (e.g., chelation therapy, etc.). Despite these drawbacks, uncertainties related to quantitative urinalysis are generally lower compared to other *in-vitro* techniques. Urinalysis is most effective when quantifying exposures of radionuclide compounds which are uniformly distributed throughout the body (e.g., tritiated water, cesium-137 ions, etc.).

2.3.2 Fecal Analysis

Radionuclide materials can also be eliminated from the body within the feces. Therefore, fecal analysis provides another means in assessing body burdens. Many of the urinalysis principles are also applicable to fecal analysis. However, radionuclides can be transferred to the feces through several different pathways compared to the singular pathway measured by urinalysis (Figure 2-1). An understanding of these pathways is necessary for interpreting fecal analysis results.

Fecal samples contain a fraction of all ingested radionuclide compounds. Insoluble compounds, which are not readily absorbed by the small intestine (e.g., radionuclide oxide forms), are almost entirely eliminated in the feces, while soluble compounds (e.g., tritiated water, ionic iodine and cesium, etc.) may be difficult to detect. Compounds present in the systemic circulation may also

appear in fecal samples through endogenous secretion (i.e., hepatobiliary secretion) into the GI tract. Finally, a portion of inhaled materials are removed from the respiratory system by mucociliary activity and are subsequently swallowed. This is the predominant respiratory elimination mechanism in the early post-exposure period (i.e., 1 to 2 days after exposure). The inhalation to fecal excretion biokinetic pathway is fairly simple to model compared to compounds which are absorbed into the systemic circulation. Therefore, fecal analysis can be very sensitive of inhalation exposures which may be difficult to detect using other *in-vitro* and *in-vivo* techniques.

The ability to detect insoluble compounds from the ingestion and/or inhalation pathways is the primary advantage of fecal analysis techniques. Also, comparing fecal and urinalysis results may provide information regarding the time of exposure. However, there are several technical and practical limitations of using fecal analyses within a routine bioassay program. These limitations include the following:

- Bioassay sensitivity is dependent on the realization of the exposure event occurrence,
- Difficulty in determining the sources of measured exposures (e.g., inhalation, ingestion, skin absorption, etc.),
- Difficulty in collecting a true 24-hour sample because of the temporal variability of fecal output,
- Inability to adjust output fluctuations to represent 24-hour samples (i.e., fecal samples lack indexing measures),
- Interference from environmental radionuclides may mask low level intakes of occupational exposures,
- and, Collection and analysis of fecal materials is more objectionable in practice compared with other bioassay procedures.

Fecal analysis will not be used for routine monitoring in the SNL/NM Internal Dosimetry Program for these reasons. However, fecal analysis may be used to evaluate inhalation exposures to insoluble radionuclide compounds resulting from an accident situation. The SNL/NM Internal Dosimetry Program maintains a capability for fecal analysis using specialized collection kits and a contract analytical laboratory.

2.3.3 Blood Analysis

The direct measurement of radionuclide materials within the systemic circulation can be made by collecting blood samples. A potential advantage of blood analysis, compared to urinalysis, is that body burdens can be estimated without the uncertainty inherent in standard renal function models. Blood samples show less fluctuations than single-void urine samples. However, radionuclide compounds may be rapidly deposited within body organs. These body burdens may not be detected by a single blood sample and may lead to incorrect exposure assessments. 24-hour blood samples would be more representative but are impractical due to the limits on the amount of blood which can be withdrawn from an individual. Consideration must also be given to the potential presence of blood-borne pathogens (e.g., hepatitis-B, etc.) which present significant biohazards to sampling and analytical personnel. For these reasons, blood analysis is not recommended for use in the SNL/NM Internal Dosimetry Program.

2.3.4 Breath Analysis

Radionuclide compounds may also be exhaled from the body. Body burdens of radium-226 and thorium-232 can be estimated using breath analysis since gaseous progeny are produced from these radionuclides. Measurement of these noble gases require specialized procedures and equipment which are not routinely available at SNL/NM. Tritiated and carbon-14 labeled compounds metabolized into tritiated water and carbon dioxide can also be detected by breath analysis. However, the collection of samples and subsequent body burden estimation of these radionuclides are more easily accomplished using other techniques (e.g., urinalysis). Routine breath analysis will not be performed in the SNL/NM Internal Dosimetry Program for these reasons. However, breath analysis remains an option to evaluate exposures from radiological incidents.

2.3.5 Nasal Smears

Nasal smears are easily collected by swiping the nasal passages adjacent to the body exterior. Deposition of radionuclide compounds in nasal passages are not well understood and detailed biokinetic models are unavailable to quantify body burdens from nasal smear samples. However, the detection of radioactive materials is useful to initiate follow-up internal dosimetry measures. Although not applicable for routine bioassay, nasal smears will be used to evaluate detected accidental inhalation exposures in the SNL/NM Internal Dosimetry Program.

2.3.6 Other *In-Vitro* Analyses

After reaching the systemic circulation, some radionuclides will eventually be incorporated in body tissues that can be readily sampled. Quantifying body burdens is generally not possible since biokinetic models have not been developed for many of these *in-vitro* techniques. However, they may be useful as an indicator of exposure. For example, facial and scalp hair analyses have been useful in detecting exposures from radon and thoron. Fingernails are also integrators of exposure for some radionuclides. Analysis of saliva and perspiration samples may reflect systemic body burdens. These alternate methods are not expected to be routinely used in the SNL/NM Internal Dosimetry Program since other *in-vitro* techniques (e.g., urinalysis) provide similar, or superior, information.

2.4 Exposure Duration

Occupational internal intakes of radionuclides can be categorized as either acute or chronic exposures, or possibly an amalgamation of the two. Acute intakes occur over a short time period and typically result from a specific event (e.g., loss of containment, etc.). Chronic intakes involve continuous exposures to radioactive materials over long periods of time (e.g., constant low-level airborne exposures, etc.). Radiological controls, facility engineering, and worker training at SNL/NM are specifically designed to detect and prevent chronic exposures. Once detected, chronic exposures are minimized, or prevented, by improving facility design (e.g., revised containment designs), procedures (e.g., develop different processes), and individual worker practices (e.g., personal protective clothing). Therefore, chronic intakes of radioactive materials are not expected to occur at SNL/NM. Internal exposures at SNL/NM are expected to result from radiological incidents (e.g., accident situations, failure to follow procedures, etc.) resulting in acute exposures which may or may not be detected. The maximum dose estimate error caused by assuming that a chronic intake is acute has been estimated to be no greater than one percent in the NCRP Report 84 *General Concepts for the Dosimetry of Internally Deposited Radionuclides* (NCRP 85). Therefore, all intakes are assumed to occur from acute events in the SNL/NM Internal Dosimetry Program.

3.0 SNL/NM Contaminants of Potential Concern

SNL/NM activities may potentially expose workers to a variety of radionuclide compounds. However, the majority of internal exposures at SNL/NM are expected to result from a relatively few radionuclide COPC. The primary COPC at SNL/NM are tritium, the isotopes of uranium, and the isotopes of plutonium. The proper selection and interpretation of bioassay techniques requires an understanding of the physical, radiological, and biologic properties of these COPC.

The deposition, absorption, distribution, retention, and elimination of radioactive materials in the body involve complex processes which can substantially differ between individuals. The 10 CFR 835 and RCM requirements are based on biokinetic models reported in International Commission on Radiological Protection (ICRP) Publication 30 *Limits for Intakes of Radionuclides by Workers* (ICRP 78). This TBD recognizes the ICRP 30 recommendations as the best practicable methodology to assess internal radionuclide exposures of large worker populations. Therefore, this TBD adopts all of the ICRP principles for use in the SNL/NM Internal Dosimetry Program. The ICRP has published revised methodologies and philosophies for monitoring internal exposures in ICRP 60 (ICRP 90a) and ICRP 61 (ICRP 90b). This TBD will be revised, as necessary, to incorporate new recommendations from these publications once they are recognized in the 10 CFR 835 and the RCM.

3.1 Tritium

Tritium (^3H or T) is the only radioactive species of the three major isotopes of hydrogen (e.g., protium (H), deuterium (D), and T). Tritium decays by beta emission with an average energy of 5.7 KeV and a radiological half-life of 12.3 years. Tritium may be encountered in numerous physical and chemical forms since its chemical properties are almost identical to H which is ubiquitous in the environment. Tritium is normally encountered at SNL/NM as tritium gas, in the form of HT, DT, or T_2 , and tritiated water, in the form of HTO, DTO, or T_2O .

3.1.1 SNL/NM Facilities Using Tritium

Radiological workers at the following facilities which process, store, or are potentially contaminated with tritium should be considered for participation in the SNL/NM Internal Dosimetry Program:

TA I: Building 891 - Facilities containing erbium tritide (ErT_3)

TA II: Entire Area - Tritium may be contained within test components which can be converted to DT or HTO when exploded.

TA III: Entire Area - Tritium is a COPC at some environmental restoration sites (e.g., Mixed Waste Landfill, etc.)

TA IV: PBFA-II and SABER facilities use accelerator targets containing tritium

TA V: ACRR, Hot Cell Facility, SPR facilities are potentially contaminated by tritium compounds.

CTF: (none known)

TTR: (none known)

3.1.2 Biokinetics of Gaseous Tritium

Tritium gas (HT) can be directly absorbed into the systemic circulation through the lungs following inhalation exposures. According to the laws of partial pressures, the absorbed HT remains in the body with a biological half-life of about two minutes (MLM 91). Most of the HT is eliminated through exhalation. However, a small percentage (e.g., 0.003 to 0.005 percent) is converted to tritiated water (HTO) within the GI tract and remains in the body for longer time periods. The fraction of HT converted to HTO behaves identically to exposures to HTO. Skin absorption of HT has been found to be negligible (MLM 91).

The DAC of HT is 2×10^{10} Bq/m³ which is 25,000 times less than the HTO DAC (i.e., 8×10^5 Bq/m³) (ICRP 78). The HT DAC is based on lung dose limits since the short residence time of HT in the body limits the whole-body exposure. In fact, the dose contribution from the HTO converted from HT is approximately equal to the lung exposure from HT. Inhalation exposures to tritium at SNL/NM is expected to involve both HT and HTO. In most applications, the exposure from HTO will be the limiting factor since its DAC is more than 4 orders of magnitude lower than the HT DAC. Therefore, bioassay techniques for HTO are considered appropriate for HT monitoring of workers participating in the SNL/NM Internal Dosimetry Program. Special bioassay procedures (e.g., breath analysis) may be performed if air sampling indicates the potential for chronic occupational exposures to HT.

3.1.3 Biokinetics of Tritiated Water

HTO can be absorbed into the body through multiple pathways. HTO is efficiently and rapidly absorbed into the systemic circulation after inhalation exposures (e.g., up to 99 percent efficient). Ingested HTO is almost entirely absorbed by the GI tract and is distributed throughout the whole-body in minutes. The skin absorption pathway can also be important due to the normal movement of water through the skin, especially in hot weather. The measured percutaneous absorption may be equivalent to approximately 50% of the HTO inhalation exposure under certain conditions (MLM 91).

Regardless of the exposure pathway, a large fraction of the absorbed HTO is uniformly distributed throughout the whole-body in the form of body water. The retention of this fraction is characteristic of the normal turn-over rate of water which can be modeled as simple exponential function. A smaller HTO fraction becomes incorporated into labile (i.e., exchangeable hydrogen bond sites) and stronger, non-labile hydrogen bond sites in organic

molecules. This fraction exhibits the longer turn-over rate characteristic of cellular components which can be modeled as the sum of two or more exponential terms. Therefore, the HTO intake retention function (IRF) can be described as follow (ICRP 78):

$$IRF(t)_{HTO} = A e^{-\lambda_1 \bullet t} + B e^{-\lambda_2 \bullet t} + C e^{-\lambda_3 \bullet t}$$

where: A = Fraction of exposure contained within body water

λ_1 = Removal rate of body water (d^{-1})

B = Fraction of exposure contained at labile molecular sites

λ_2 = Turnover rate of labile cellular components (d^{-1})

C = Fraction of exposure contained at non-labile molecular sites

λ_3 = Turnover rate of non-labile cellular components (d^{-1})

t = post acute exposure time interval (d)

IRF(t)_{HTO} = Fraction of initial HTO exposure retained at time t

The overall dose contribution from labile and non-labile site HTO depositions have been found to represent less than 10 percent of the CEDE (ICRP 78). The ICRP does not include the organically bound tritium fraction in the HTO ALI calculation. Therefore, estimates of CEDE from HTO exposures during routine bioassay monitoring in the SNL/NM Internal Dosimetry Program will include only the body water component. The removal rate of HTO in body waters can be calculated as follows:

$$\lambda_1 = \frac{0.693}{T_R} + \frac{0.693}{T_B}$$

where: T_R = Radiological half-life of tritium (d)

T_B = Biological clearance half-life of body water (d)

λ_1 = Removal rate of HTO in body water (d^{-1})

The biological clearance half-life of body water (T_B) is closely related to fluid intake patterns. Individual patterns can fluctuate widely depending on such factors as ambient air temperature, perspiration rates, and personal lifestyles (e.g., exercise patterns). Values of T_B have been observed to range between 4 and 18 days with an average of about 10 days. The average value is in agreement with the approximate half-life obtained based on standard man assumptions (i.e., Daily water intake rate of 3,000 grams and a constant 42,000 grams total body water mass). Therefore, the T_B value for body waters will be assumed to be 10 days for routine applications of the SNL/NM Internal Dosimetry Program. Inserting the T_B (i.e, 10 days) and T_R (i.e. 4490 days) values yields the following HTO IRF:

$$IRF(t)_{HTO} = \frac{C_t}{C_o} = e^{-0.0695 \bullet t}$$

where: C_t = Concentration of HTO in body water at time t

C_o = Initial concentration of HTO in body water

t = time post acute exposure (d)
 $IRF(t)_{HTO}$ = Retention of HTO

HTO is assumed to be totally eliminated from the body in urine. Therefore, urine HTO concentrations is representative to body water HTO concentrations. If necessary, radiological incident evaluations (e.g., accident evaluations) may include the organically bound components in the HTO retention function for more precise CEDE estimates.

3.1.4 Biokinetics of Other Tritiated Compounds

Tritiated vacuum pump oil may be a significant exposure source at SNL/NM accelerator facilities. Tritium specific activities may range between a few mCi/L to a few tens of Ci/L (MLM 91). The primary radiological hazard from tritiated pump oils remains HTO since it has greater skin permeability. Routine bioassay monitoring of HTO should be sufficient to monitor exposures to tritiated oils (MLM 91).

Tritium may also be encountered in the form of metal tritides (e.g., titanium tritide and erbium tritide) during neutron generator research at SNL/NM. Current health protection for guidelines for metal tritides are based on the assumption that the biological behavior of these compounds is similar to HTO. However, recent research suggests that retention times of metal tritides may significantly exceed HTO values (SNL ?). The SNL/NM Internal Dosimetry Program will continue to use the HTO models to represent metal tritides until new models are developed and confirmed by animal tests.

Another source of HTO contamination is tritiated solvents. Exposure pathways from solvents include skin absorption and inhalation of the volatile components. Solvents may be deposited in organs which are not normally effected by HTO. The limiting consideration of these exposures may be the chemical toxicity of the solvent. Exposure assessment techniques are expected to be variable depending on the solvent form. Facilities using tritiated solvents will require site-specific consideration in the SNL/NM Internal Dosimetry Program.

Skin contact with tritium contaminated glass or metal surfaces have been shown to result in exposures to organic tritium forms. Such exposures can result in elevated tritium concentrations at the exposure site and other tissues in addition to large amounts of organic tritium in urine. The biokinetics of this process are poorly understood, but research does suggest that the skin dose at the point of contact may contribute the most significant dose (MLM 91). The tritium dose contribution from these compounds are often insignificant compared to the concurrent absorbed HTO dose. Therefore, surface contamination sources which generate organic tritiated compound exposures will not be considered during routine bioassay monitoring in the SNL/NM Internal Dosimetry Program. Individual exposure assessments following radiological incidents may include consideration of these exposures.

3.1.5 Techniques and Considerations for Tritium Bioassay

In-vivo bioassay techniques to assess T body burdens are not possible since the beta emission can not be detected outside of the body. Tritium is quickly distributed throughout the whole-body within 2 hours post exposure to HTO. Once equilibrium is achieved, the body burden of T can be assessed using one of several *in-vitro* bioassay techniques (e.g., blood, saliva, and urine). Fecal analyses is not practiced for monitoring tritium exposures since HTO is assumed to be totally absorbed from the GI tract. Because the collection and interpretation is relatively simple, urinalysis will be used to assess T body burdens in the SNL/NM Internal Dosimetry Program.

Urine samples collected shortly after exposure may not have had sufficient time to establish equilibrium within body waters. Subsequent analysis of these samples would not be representative of the true T body burden. The preferred method is to wait 2 hours followed by voiding the non-representative urine sample. This sample may be used to represent pre-accident, baseline conditions (MLM 91).

3.1.6 Tritium Intake Limits

Because a critical organ has not been identified, the ingestion and inhalation pathway ALI for HTO are determined from the stochastic limit to the whole-body (i.e., 5 rem CEDE limit). The ALI for HTO skin absorption and wound exposures are also assumed to be equivalent to these values. Table 3-1 summarizes the ALI and the DAC for exposures to HT and HTO (ICRP 79):

Physical/Chemical Form	Parameter	Ingestion Pathway (Limiting Factor)	Inhalation Pathway (Limiting Factor)
Tritiated Water (HTO)	ALI (Bq) ¹	3 x10 ⁹ (Stochastic)	3 x10 ⁹ (Stochastic)
	DAC (Bq/m ³) ²	N/A	8 x10 ⁵
Elemental Tritium (HT)	ALI (Bq)	N/A	N/A
	DAC (Bq/m ³)	N/A	2 x10 ¹⁰ (Non-stochastic - Lungs)

¹ 1 Bq = 2.7 x10⁻⁵ uCi

² 1 Bq/m³ = 27 uCi/ml

Table 3-1 Tritium Intake Limits

3.2 Uranium

Isotopes of uranium may be encountered at several facilities or activities at SNL/NM. However, occupational exposures mainly occur from natural uranium (U-Nat), depleted uranium (DU), and enriched uranium (EU) containing different proportions of the naturally occurring

uranium isotopes (i.e., ^{234}U , ^{235}U , and ^{238}U). Typical isotopic abundances by weight and activity are displayed in Table 3-2 (EGG 88):

Isotope	Weight % U-Nat	Weight % DU	Weight % EU*	Activity % U-Nat	Activity % DU	Activity % EU*
^{238}U	99.2739	99.75	97.01	0.473	0.901	0.144
^{235}U	0.7204	0.25	2.96	0.022	0.015	0.028
^{234}U	0.0057	0.0005	0.03	0.505	0.084	0.828

* Typical Commercial Feed Enrichment

Table 3-2: Compositions of Materials Containing Uranium at SNL/NM

The isotopic abundances for DU and EU in Table 3-2 are applicable for uranium compounds processed using gaseous diffusion. Uranium processing can also be performed using laser isotope separation techniques which would generate different isotopic ratios . However, most uranium materials encountered at SNL/NM are anticipated to be the result of gaseous diffusion processing.

Progeny of these isotopes are radioactive and form decay chains. Uranium-238 and ^{234}U are members of the uranium decay series, while ^{235}U is a member of the actinium decay series. Several of these progeny may have significant internal dosimetry implications when secular equilibrium is maintained. However, most of the uranium forms encountered at SNL/NM have been chemically extracted from the virgin feed materials. Progeny with long half-lives (e.g., ^{230}Th in the uranium series, and ^{231}Pa in the actinium series) effectively prevent secular equilibrium in these cases. The resultant radionuclide progeny which occur in significant abundance to impact radiological controls are ^{234}Th and $^{234\text{m}}\text{Pa}$ in the uranium series, and ^{231}Th in the actinium series (EGG 88). However, other decay progeny may be present from incomplete chemical separations and from naturally occurring deposits of uranium. Tables 3-3 and 3-4 present the radiological characteristics of the principle uranium COPC at SNL/NM:

Uranium Decay Series Member	Radionuclide Half-Life	Energies (MeV) and % Abundances of Major Emissions		
		Alpha	Beta	Gamma
Uranium-238	4.51 x10 ⁹ y	4.15 (25%) 4.20 (75%)	(none)	(none)
Uranium-234	2.47 x10 ⁵ y	4.72 (28%) 4.77 (72%)	(none)	0.053 (0.2%)
Thorium-234	24.1 d	(none)	0.103 (21%) 0.193 (79%)	0.063 (3.5%) 0.093 (4%)
Proactinium-234	1.17m	(none)	2.29 (98%)	0.765 (0.3%) 1.001 (0.6%)

Table 3-3: Significant Uranium Decay Series COPC at SNL/NM

Actinium Decay Series Member	Radionuclide Half-Life	Energies (MeV) and % Abundances of Major Emissions		
		Alpha	Beta	Gamma
Uranium-235	7.04 x10 ⁸ y	4.37 (18%) 4.40 (57%) 4.58 (8%)	(none)	0.144 (11 %) 0.185 (54 %) 0.204 (5%)
Thorium-231	25.5 h	(none)	0.14 (45%) 0.22 (15%) 0.305 (40%)	0.026 (2%) 0.084 (10%)

Table 3-4: Significant Actinium Decay Series COPC at SNL/NM

All of the uranium isotopes decay by alpha emission while the non-uranium decay progeny are primarily beta emitters. Thus, significant internal doses result from exposures to the uranium isotopes while the dose contribution from the decay series progeny are often ignored. *In-vivo* bioassays are possible due to the presence of the uranium progeny, with the exception being ²³⁵U. The biokinetics of uranium compounds are dependent on the exposure pathway (e.g., inhalation, ingestion, etc.) and on the chemical solubility of the compound in bodily fluids.

3.2.1 SNL/NM Facilities Using Uranium Compounds

Radiological workers at the following facilities which process, store, or are potentially contaminated with uranium compounds should be considered for participation in the SNL/NM Internal Dosimetry Program:

TA I: (none known)

TA II: (none known)

TA III: Entire Area - Uranium compounds are COPC at some environmental restoration sites

TA IV: (none known)

TA V: ACRR, Hot Cell Facility, SPR facilities process or are contaminated by uranium compounds.

CTF: Entire Area - Uranium compounds are COPC at some environmental restoration sites

TTR: Entire Area - Uranium compounds are COPC at some environmental restoration sites

3.2.2 Biokinetics of Inhaled Uranium Compounds

The principle exposure pathway for uranium compounds is inhalation in most occupational situations. The deposition patterns within the respiratory system is dependent on the size of the inhaled particle. The ICRP has developed a lung model which describes deposition patterns as a function of the activity median aerodynamic diameter (AMAD) of the particle size distribution. This lung model consists of 10 mathematical compartments which approximate the biokinetics of the three principle regions of the respiratory system (i.e., nasal-pharyngeal, tracheo-bronchial, and pulmonary regions). Like all ICRP 30 models, these compartments may or may not represent a real structured physiological entity in the body. Retention within the respiratory tract is a function of the solubility of the inhaled compounds. Class Y compounds have lung retention half-lives exceeding 100 days, Class W between 10 and 100 days, and Class D less than 10 days.

Table 3-5 contains the clearance classes for uranium compounds established by the ICRP (ICRP 88):

Uranium Compound	Pulmonary Clearance Class
UF ₆ , UO ₂ F ₂ , UO ₂ (NO ₃) ₂	D
UO ₃ , UF ₄ , UCl ₄	W
UO ₂ , U ₃ O ₈	Y

Table 3-5 ICRP Pulmonary Clearance Classes for Uranium Compounds

In general, the biokinetics of inhaled particles are characterized by an initial rapid clearance, due to mucociliary transport into the GI system, followed by long-term clearance attributed to particle dissolution and mucociliary actions. Table 3-6 illustrates the ultimate biokinetic fate of inhaled 1 micron AMAD particles containing uranium based on ICRP models (EGG 88):

Clearance Class	Percent Exhaled from the body	Percent Absorbed into Systemic Circulation	Percent Removed to the GI Tract
D	37	47.6	15.4
W	37	12	51
Y	37	5.4	57.6

Table 3-6 Biokinetic Fate of 1 Micron AMAD Uranium Compounds

Although the mathematics describing single compartment transfer are relatively simple (e.g., tritium biokinetics), several computer programs have been developed to interpret the complex retention and elimination interactions of the respiratory system as a whole. The computer program *INDOS* is based on the ICRP 30 models and is used in the SNL/NM Internal Dosimetry Program to describe the biokinetics of inhaled and ingested compounds. *INDOS* can be used to describe pulmonary retention for use in *in-vivo* bioassay, and interpreting *in-vitro* bioassay results based on respiratory elimination patterns.

3.2.3 Biokinetics of Ingested Uranium Compounds

Depending on its solubility, ingested uranium compounds can be absorbed into the body via the small intestine or eliminated from the body within fecal materials. The ICRP has developed a GI Tract model which describes this process and consists of four transfer compartments. Table 3-7 provides the ICRP GI absorbed fraction (f_1) which is related to the clearance class of the uranium compound (ICRP 30):

Clearance Class	Absorbed Fraction of Activity into Systemic Circulation (f_1)
D	0.05
W	0.05
Y	0.002

Table 3-7: ICRP Absorption Fractions (f_1) for Ingestion Exposures

Uranium materials cleared from the respiratory tract by mucociliary action are also adequately modeled by the ICRP 30 GI Tract model. A maximum of 5 percent of the total ingested uranium is expected to reach systemic circulation. The remaining portion remains in the GI tract and is ultimately eliminated within fecal materials. Since there are relatively few penetrating radiation emissions from the uranium compounds (Tables 3-3 and 3-4), internal dosimetry implications of the unabsorbed portion are minimal. The *INDOS* computer program is based on the ICRP 30 GI Tract Model and accurately models ingestion exposures from uranium compounds. Therefore,

analytical results from fecal samples can be used to determine uranium body burdens using the *INDOS* program.

3.2.4 Biokinetics of Absorbed Uranium Compounds

Once introduced into the systemic circulation, via inhalation, ingestion, skin absorption, or wound exposures, all uranium compounds are assumed to be transported and retained similarly in the various body organs. Approximately 20 percent of absorbed uranium compounds are deposited in the skeleton, 12 percent within the kidneys, and 12 percent distributed throughout the whole body. Uranium IRF of various body organs are described as follows (ICRP 78):

$$IRF_B = 0.2 e^{\left(\frac{-0.693 \bullet t}{20}\right)} + 0.023 e^{\left(\frac{-0.693 \bullet t}{5000}\right)}$$

$$IRF_K = 0.12 e^{\left(\frac{-0.693 \bullet t}{6}\right)} + 0.00052 e^{\left(\frac{-0.693 \bullet t}{1500}\right)}$$

$$IRF_O = 0.12 e^{\left(\frac{-0.693 \bullet t}{6}\right)} + 0.00052 e^{\left(\frac{-0.693 \bullet t}{1500}\right)}$$

where: t = Time post acute intake (d)

IRF_B = Bone uranium retention function

IRF_K = Kidney uranium retention function

IRF_O = All other body organs uranium retention function

The balance of the uranium compounds absorbed into the systemic circulation (i.e., approximately 54%) is assumed to be directly eliminated from the body within urine. The IRF for this component is defined as follows (ICRP 88):

$$IRF_U = 0.54 e^{\left(\frac{-0.693 \bullet t}{0.25}\right)}$$

where: t = Time post acute intake (d)

IRF_U = retention function for fast urinary clearance fraction

All uranium removed from these mathematical compartments is assumed to be eliminated within urine. Therefore, urinalysis can be used to determine uranium systemic body burdens using the *INDOS* program. The bone uranium retention function is of additional interest since the bone surface is considered the critical organ from uranium exposures. The kidney retention function is useful in determining kidney uranium burdens for comparison with regulatory limits concerning uranium nephrotoxicity.

3.2.5 Chemical Toxicity of Uranium Compounds

Health effects from exposures to uranium compounds include chemical toxicity hazards (i.e., nephrotoxicity) in addition to radiological hazards. In certain situations, acceptable radiological exposures may exceed regulatory thresholds for chemical exposures. The determination of the limiting hazard (i.e., radiological or chemical) is dependent on the solubility of the uranium compound, the isotope enrichment, and the duration of exposure (EGG 88). Based on a radiological limit of 5 rem and chemical limits derived from the Occupational Safety and Health Administration (OSHA) standards, chemical toxicity effect thresholds are more limiting for exposures to DU and U-Nat compounds of transportability class D and W (EGG 88). In contrast, class Y compounds are always limited by radiological concerns, regardless of the uranium enrichment. The majority of uranium exposures at SNL/NM are expected to be in the form of oxides (i.e., class Y chemical forms). Therefore, the governing regulatory constraints are expected to be based on radiological hazards. However, the SNL/NM Industrial Hygiene Department (7712) will be notified when uranium exposures are detected in the SNL/NM Internal Dosimetry Program to monitor uranium chemical toxicity concerns.

3.2.6 Techniques and Considerations of Uranium Bioassay

In-vivo bioassay of uranium exposures are possible due to the radiation emissions of uranium progeny and of ^{235}U (Tables 3-3 and 4-3). However, the SNL/NM Radiation Protection Measurement Department (7715) WBC is not considered sufficiently sensitive to detect small uranium body burdens. Therefore, the Department 7715 WBC will not be used for routine bioassays. More sensitive *in-vivo* bioassay techniques, such as lung counting, can be performed at a contract laboratory (i.e., LANL) to evaluate significant exposures to these compounds. It should be noted that bioassays based on uranium progeny assume that the progeny is retained at the same location within the body as the parent isotope.

Table 3-6 demonstrates the potential usefulness of fecal analysis for *in-vitro* bioassay. These assays can be performed in the event of an identified significant exposure event. However, routine monitoring is not recommended in the SNL/NM Internal Dosimetry Program since uranium compounds are rapidly removed via feces and may not be detected. Thus, urinalysis will be the *in-vitro* technique of choice for routine monitoring applications in the SNL/NM Internal Dosimetry Program.

The uranium isotopes are ubiquitous throughout the environment. Non-occupational exposures occur from dietary uranium intake through foods and drinking water. As a result, workers routinely ingest approximately 2 micrograms of uranium each day which may be misinterpreted-interpreted as occupational exposures (ICRP 79). Because these body burdens result from chronic exposures which result in fairly uniform elimination rates, baseline bioassays (i.e., urinalysis) are invaluable to discriminate occupational and environmental exposures. Special bioassay studies of non-radiological workers may prove useful in determining average environmental uranium body burdens of SNL/NM workers. For the purposes of this TBD, chronic uranium exposures from food and water consumed at SNL/NM are considered part of

the worker's environmental exposure (i.e., all occupational exposures to uranium result from acute exposure events).

The time of *in-vitro* bioassay sample collection may be useful in determining the source of exposure. Collections at the end of the work week would be more sensitive for occupational exposures, while samples collected after extended leaves of absence (e.g., vacations) would result in more accurate assessments of environmental exposures.

3.2.7 Intake Limits of Uranium COPC

The ALI of uranium compounds which are quickly absorbed into the systemic circulation (e.g., inhalation of class D compounds, ingestion of class D and W compounds, and wound exposures) are determined from the non-stochastic dose limit (i.e., 50 rem) to the bone surfaces. The ALI from other exposures are limited by the stochastic dose limit to the whole-body. Tables 3-8 through 3-10 summarizes the ALI and the DAC for exposures to ^{234}U , ^{235}U , and ^{238}U (ICRP 79):

^{234}U Pulmonary Clearance Class	Parameter	Ingestion Pathway (Limiting Factor)	Inhalation Pathway ¹ (Limiting Factor)
D	ALI (Bq) ²	4×10^5 (Non-Stochastic - bone surfaces)	5×10^4 (Non-stochastic - bone surfaces)
	DAC (Bq/m ³) ³	N/A	2×10^1
W	ALI (Bq)	4×10^5 (Non-stochastic - bone surfaces)	3×10^4 (Stochastic)
	DAC (Bq/m ³)	N/A	1×10^1
Y	ALI (Bq)	7×10^6 (Stochastic)	1×10^3 (Non-stochastic)
	DAC (Bq/m ³)	N/A	6×10^{-1}

¹ ALI and DAC from Inhalation Exposures Assume a 1 micron AMAD

² 1 Bq = 2.7×10^{-5} uCi

³ 1 Bq/m³ = 27 uCi/ml

Table 3-8 ICRP Intake Limits of Uranium-234

²³⁵ U Pulmonary Clearance Class	Parameter	Ingestion Pathway (Limiting Factor)	Inhalation Pathway ¹ (Limiting Factor)
D	ALI (Bq) ²	5 x10 ⁵ (Non-Stochastic - bone surfaces)	5 x10 ⁴ (Non-stochastic - bone surfaces)
	DAC (Bq/m ³) ³	N/A	2 x10 ¹
W	ALI (Bq)	5 x10 ⁵ (Non-stochastic - bone surfaces)	3 x10 ⁴ (Stochastic)
	DAC (Bq/m ³)	N/A	1 x10 ¹
Y	ALI (Bq)	7 x10 ⁶ (Stochastic)	2 x10 ³ (Non-stochastic)
	DAC (Bq/m ³)	N/A	6 x10 ⁻¹

¹ ALI and DAC from Inhalation Exposures Assume a 1 micron AMAD

² 1 Bq = 2.7 x10⁻⁵ uCi

³ 1 Bq/m³ = 27 uCi/ml

Table 3-9 ICRP Intake Limits of Uranium-235

²³⁸ U Pulmonary Clearance Class	Parameter	Ingestion Pathway (Limiting Factor)	Inhalation Pathway ¹ (Limiting Factor)
D	ALI (Bq) ²	5 x10 ⁵ (Non-Stochastic - bone surfaces)	5 x10 ⁴ (Non-stochastic - bone surfaces)
	DAC (Bq/m ³) ³	N/A	2 x10 ¹
W	ALI (Bq)	5 x10 ⁵ (Non-stochastic - bone surfaces)	3 x10 ⁴ (Stochastic)
	DAC (Bq/m ³)	N/A	1 x10 ¹
Y	ALI (Bq)	8 x10 ⁶ (Stochastic)	2 x10 ³ (Non-stochastic)
	DAC (Bq/m ³)	N/A	7 x10 ⁻¹

¹ ALI and DAC from Inhalation Exposures Assume a 1 micron AMAD

² 1 Bq = 2.7 x10⁻⁵ uCi

³ 1 Bq/m³ = 27 uCi/ml

Table 3-10 ICRP Intake Limits of Uranium-238

All of the uranium isotopes commonly encountered at SNL/NM have similar ALI with uranium-234 being the most restrictive. Therefore, exposures to uncharacterized mixtures of uranium

isotopes (e.g., DU, EU, U-Nat) can be adequately evaluated by assuming exposure to ^{234}U only. The most common pulmonary clearance class for uranium compounds typically encountered at SNL/NM is assumed to be Class Y.

3.3 Plutonium

Plutonium exposures at SNL/NM are expected to be rare compared to other exposure sources. The biokinetics of plutonium isotopes are included in the SNL/NM COPC because of the high degree of public sensitivity to these compounds and their low ALI (i.e., high radiotoxicity). The principle isotopes of plutonium found in non-production DOE facilities are ^{238}Pu and ^{239}Pu . The radiological properties of these isotopes are summarized in Table 3-11. Compounds of these isotopes potentially contain other radionuclides. These contaminants are typically found in minute quantities and are subsequently ignored in routine bioassay programs. The exception is americium-241, which is significant for *in-vivo* bioassay application and is also noted in Table 3-11 (PNL 88).

Isotope	Radionuclide Half-Life	Energies (MeV) and % Abundances of Major Emissions		
		Alpha	X-ray	Gamma
Plutonium-238	87.7 y	5.50 (71.6) 5.46 (28.3)	0.011 to 0.021 (10.5)	(none)
Plutonium-239	2.41×10^4 y	5.156 (73.8) 5.143 (15.2) 5.105 (10.7)	0.0116 to 0.0215 (4.8)	(none)
Americium-241	432 y	5.486 (85.2) 5.443 (12.8) 5.389 (1.4)	0.0119 to 0.0222 (37.6)	0.0595 (35.7) 0.0263 (2.4)

Table 3-11: Plutonium COPC at SNL/NM

All of the SNL/NM plutonium COPC decay by alpha particle emission which cannot be detected using normal *in-vivo* techniques. However, *in-vivo* bioassay is possible by assessing the uranium L x-rays from plutonium decay. Detection of these x-rays are difficult due to their low energies which can be easily attenuated within the subject. The 59.5 KeV gamma ray emission from ^{241}Am is easier to detect and can be used to quantify smaller exposures of plutonium when the $^{241}\text{Am}/\text{Pu}$ ratio is known. The biokinetics of plutonium compounds are dependent on the exposure pathway (e.g., inhalation, ingestion, etc.) and on the chemical solubility of the compound in body fluids.

3.3.1 SNL/NM Facilities Using Plutonium

Radiological workers at the following facilities which process, store, or are potentially contaminated with plutonium compounds should be considered for participation in the SNL/NM Internal Dosimetry Program:

TA I: (none known)

TA II: (none known)

TA III: (none known)

TA IV: (none known)

TA V: ACRR, Hot Cell Facility, SPR facilities process or are contaminated by plutonium compounds.

CTF: (none known)

TTR: Entire Area - plutonium compounds are COPC at some environmental restoration sites.

3.3.2 Biokinetics of the Plutonium Exposure Pathways

Deposition patterns of inhaled plutonium compounds are assumed to adequately described by the ICRP Lung model (ICRP 86). Retention in the lungs is a function of the compound's chemical solubility in body fluids. Table 3-12 contains the pulmonary clearance class for plutonium compounds (ICRP 86).

Compound Type	Pulmonary Clearance Class
Plutonium (none known) All except oxides Oxides	 D W Y
Americium All Compounds	 W

Table 3-12 Pulmonary Clearance Classes of Americium and Plutonium Compounds

Ingestion exposures are typically insignificant due to the low absorption rate of plutonium compounds into systemic circulation. The ICRP GI tract model is assumed to adequately model

plutonium ingestion exposures. Table 3-13 contains the f_1 fractions for the plutonium COPC (PNL 88).

Compound Type	f_1
Plutonium	
Oxides (except "polydisperse" oxides)	0.00001
Nitrates	0.0001
Other compounds or unknown mixtures	0.001
Americium	
All Compounds	0.001

Table 3-13 GI Absorption Fractions for Plutonium Compounds

The skin absorption pathway is also considered insignificant due to the low solubility of plutonium compounds. Wound exposures often result in the most significant plutonium exposures at DOE facilities. All of the plutonium compound is assumed to be directly introduced into systemic circulation unless proven otherwise by wound counting *in-vivo* bioassay techniques.

3.3.3 Biokinetics of Absorbed Plutonium Compounds

Once introduced into the systemic circulation, plutonium is primarily deposited in the liver and skeletal system (PNL 88). The reported distribution and the biologic clearance times varies between ICRP 30 and ICRP Publication 48 *The Metabolism of Plutonium and Related Elements* (ICRP 86). The ICRP 30 distribution factors are used in conjunction with the biologic clearance parameters from ICRP 48 to model plutonium retention in the ICRP Publication 54 *Individual Monitoring for Intakes of Radionuclides by Workers: Design and Interpretation* (ICRP 88). The SNL/NM Internal Dosimetry Program adopts this approach since this philosophy reflects the most current guidance from the ICRP. The following IRF for the bone and liver compartments are derived using the ICRP 54 recommendations:

$$IRF_B = 0.45 e^{\left(\frac{-0.693 \bullet t}{18250}\right)}$$

$$IRF_L = 0.45 e^{\left(\frac{-0.693 \bullet t}{7300}\right)}$$

where: IRF_B = Bone intake retention function

IRF_L = Liver intake retention function

t = time post acute exposure (d)

The remaining 10 percent of the systemic body burden is assumed to be distributed to a number of minor deposition sites. The ICRP considers the conservatism built into the liver and bone models (i.e., dose over-estimation) compensates for ignoring the minor deposition sites in

dosimetry evaluations (ICRP 86) However, dose assessments to the gonads may be of interest since the biological clearance time of these organs is considered infinite. Partitioning factors vary from 0.035 percent in males, to 0.011 percent in females. The partitioning value for males with a biological clearance half-time of 100 years will be conservatively used for routine bioassay applications in the SNL/NM Internal Dosimetry Program. The remaining fraction of plutonium in systemic circulation is modeled by urine elimination with a biological half-life of 0.25 days. The adoption of these assumptions results in remaining organ-specific IRF for absorbed plutonium:

$$IRF_G = 0.00035 e^{\left(\frac{-0.693 \cdot t}{36500}\right)}$$

$$IRF_U = 0.09965 e^{\left(\frac{-0.693 \cdot t}{0.25}\right)}$$

where: IRF_G = Gonad retention function

IRF_U = Early urinary excretion retention function

t = times post acute intake (d)

The plutonium burdens of these organs can be derived from intake estimates using the *INDOS* computer program. In addition, these equations provide an accurate description of total daily plutonium excretion (i.e., urine and fecal output) at 10,000 days post exposure or greater (Stannard 88). However, these IRF are inadequate in predicting excretion patterns 5 years (1825 d) or earlier post exposure (Stannard 88). Since the goal of the SNL/NM Internal Dosimetry Program is early detection of internal exposures, alternate methods based on excretion models are required to estimate intakes from plutonium *in-vitro* bioassay data.

3.3.4 Excretion Patterns of Plutonium Compounds

Plutonium excretion models were developed from human experimental data and the monitoring of known occupational exposures. Initial models based on power functions were derived by Langham as follows (Stannard 88):

$$E_U = 0.2 t^{-0.74}$$

$$E_F = 0.63 t^{-1.09}$$

where: E_U = Percent of original intake of plutonium in urine at time t

E_F = Percent of original intake of plutonium in feces at time t

t = time post exposure (d), t > 1

The Langham equations were re-evaluated in the 1970s and were found to be accurate in predicting plutonium excretion up to 5 years post exposure. However, the power function form of these equations is not compatible with the *INDOS* program which is based on exponential functions. Fortunately, exponential excretion functions were developed which adequately fits the earlier plutonium data. The Jones and Durbin equations describes the excretion of plutonium in urine and feces as follows (Jones 85, Durbin 72):

$$E_U E_F = 0.000075 e^{(-0.5658 \bullet t)} + 0.000239 e^{(-0.05442 \bullet t)} + \\ 0.000083 e^{(-0.012938 \bullet t)} + 0.0000142 e^{(-0.000284 \bullet t)} + \\ 0.000012 e^{(-0.000173 \bullet t)}$$

where: E_U = Fraction of original intake of plutonium in urine at time t

E_F = Fraction of original intake of plutonium in feces at time t

t = time post exposure (d)

These equations are considered reliable for predicting excretion up to 5 years post exposure.

Initial intakes will be over-estimated when evaluating later excretion data using these equations (Stannard 88). The excretion function are related to the IRF by the following relationship:

$$E(t)_i = -f_i \bullet \frac{d}{dt} IRF(t)$$

where: f_i = Fraction of systemic burden secreted by route i

$IRF(t)$ = The systemic intake retention function

E_i = Daily excretion function of route i

Therefore, a pseudo systemic IRF can be derived by integrating the excretion function and defining the excretion fraction for the respective elimination routes. The "pseudo" designation is important since the IRF inferred does not represent actual biokinetic compartments (i.e., equation is based on observed excretion patterns, not systemic uptake patterns). However, the pseudo IRF can be used to estimate the initial intake of plutonium compounds using the *INDOS* program. The derived pseudo IRF equations are as follows:

$$IRF_{UF} = 0.00159 e^{(-0.23558 \bullet t)} + 0.0244 e^{(-0.10842 \bullet t)} + \\ 0.00791 e^{(-0.012938 \bullet t)} + 0.08932 e^{(-0.000284 \bullet t)} + \\ 0.5663 e^{(-0.000173 \bullet t)}$$

where: IRF_U = Pseudo IRF for determining intake from urinalysis

IRF_F = Pseudo IRF for determining intake from fecal analysis

t = time post acute intake (d)

The excretion fractions for use in the *INDOS* program are as follows (Jones 85, and Durbin 72):

Plutonium fraction excreted as urine (f_U) = 0.54

Plutonium fraction excreted as feces (f_F) = 0.12

The use of the pseudo IRF in the *INDOS* program will allow assessment of the initial uptake of plutonium compounds from fecal and urine *in-vitro* analyses.

3.3.5 Biokinetics of Absorbed Americium Compounds

The biokinetics of compounds containing ^{241}Am are considered to be similar to plutonium compounds by the ICRP (ICRP 88). Therefore, the IRF and pseudo IRF applicable for plutonium intakes will be used to evaluate ^{241}Am exposures in the SNL/NM Internal Dosimetry Program.

3.3.6 Techniques and Considerations in Plutonium Bioassay

In-vivo bioassay of plutonium exposures require specialized facilities to detect and quantify the low-energy x-ray emissions from plutonium isotopes. The sensitivity of the SNL/NM WBC is insufficient to detect significant plutonium exposures (e.g., exposures exceeding the ALI). Therefore, *in-vivo* plutonium bioassay is not practical for routine monitoring at SNL/NM. Lung counting facilities at LANL will be used when inhalation exposures to plutonium compounds are suspected.

Routine *in-vitro* bioassay for plutonium exposures at SNL/NM will be primarily performed by urinalysis. Fecal analyses can also be performed to characterized suspected exposures to plutonium. However, routine fecal monitoring is not recommended in the SNL/NM Internal Dosimetry Program since inhaled plutonium compounds are rapidly cleared from the body and may not be detected.

Because of the small ALI for plutonium isotopes, the analytical sensitivity achieved for *in-vitro* analyses is not sufficient to meet the monitoring requirements in 10 CFR 835 (i.e., detect exposures equivalent to 100 mRem CEDE). In addition, the bioassay frequency necessary to meet these requirements may be prohibitively frequent for worker convenience and cost considerations. Therefore, plutonium bioassays will be supplemented with air sampling data from the SNL/NM Air Sampling Program since these results may be more sensitive in detecting plutonium exposures. The use of personal air samplers are highly recommended whenever significant exposures to plutonium are possible.

3.3.7 Intake Limits of Plutonium COPC

The ALI of plutonium and americium compounds are determined from the non-stochastic dose limit (i.e., 50 rem) to the bone surfaces except for class Y ^{238}Pu which is based on stochastic limits (ICRP 88). Tables 3-14 through 3-16 summarizes the ALI and the DAC for exposures to ^{238}Pu , ^{239}Pu , and ^{241}Am (ICRP 79 and ICRP 88):

^{238}Pu Pulmonary Clearance Class	Parameter	Ingestion Pathway ¹	Inhalation Pathway ²
W ($f_1 = 0.0001$)	ALI (Bq)	3×10^5	3×10^2

	DAC (Bq/m ³)	N/A	1 x 10 ⁻¹
Y (f ₁ = 0.00001)	ALI (Bq)	3 x 10 ⁶	7 x 10 ²
	DAC (Bq/m ³)	N/A	3 x 10 ⁻¹

¹ Ingestion ALI from ICRP 30

² Inhalation ALI and DAC from ICRP 54 and Assume a 1 micron AMAD

Table 3-14 ICRP Intake Limits of Plutonium-238

²³⁹ Pu Pulmonary Clearance Class	Parameter	Ingestion Pathway ¹	Inhalation Pathway ²
W (f ₁ = 0.0001)	ALI (Bq)	2 x 10 ⁵	2 x 10 ²
	DAC (Bq/m ³)	N/A	1 x 10 ⁻¹
Y (f ₁ = 0.00001)	ALI (Bq)	2 x 10 ⁶	6 x 10 ²
	DAC (Bq/m ³)	N/A	3 x 10 ⁻¹

¹ Ingestion ALI from ICRP 30

² Inhalation ALI and DAC from ICRP 54 and Assume a 1 micron AMAD

Table 3-15 ICRP Intake Limits of Plutonium-239

²⁴¹ Am Pulmonary Clearance Class	Parameter	Ingestion Pathway ¹	Inhalation Pathway ²
W (f ₁ = 0.00005)	ALI (Bq)	5 x 10 ⁴	2 x 10 ²
	DAC (Bq/m ³)	N/A	1 x 10 ⁻¹

¹ Ingestion ALI from ICRP 30

² Inhalation ALI and DAC from ICRP 54 and Assume a 1 micron AMAD

Table 3-16 ICRP Intake Limits of Americium-241

The ALI for Class W ²³⁹Pu and ²⁴¹Am are limiting. Therefore, potential exposures to unknown mixtures of plutonium COPC will be assumed to be Class W ²³⁹Pu.

3.4 Other SNL/NM COPC

Although less frequent, occupational exposures may result from exposures to additional radionuclides not specifically addressed in this TBD. Appendix B provides brief descriptions of the properties, biokinetics, and dosimetry of these COPC at SNL/NM. The bioassay philosophies used to monitor tritium, uranium, and plutonium exposures are anticipated to be adequate in monitoring exposures to other COPC at SNL/NM. Exceptions will be addressed, as needed, in site-specific TBD in the SNL/NM Internal Dosimetry Program.

The ICRP has developed IRF for alkaline earth elements (e.g., calcium, strontium, barium, and radium) which are in the following form (ICRP 72):

$$IRF(t) = (1 - p)e^{-mt} + p \varepsilon^b (t + \varepsilon)^{-b} [\beta e^{-r\lambda t} + (1 - \beta)e^{-\sigma\lambda t}]$$

Such expressions can not be directly used by the *INDOS* computer program. However, exponential fits which approximate the ICRP functions have been developed in (**Reference**). Appendix B contains the exponential IRF forms which will be used to evaluate exposures from alkaline earth elements in the SNL/NM Internal Dosimetry Program.

4.0 Bioassay Participation Requirements and Frequencies

Radiological workers may be required to participate in either routine or special bioassay programs in the SNL/NM Internal Dosimetry Program. The routine bioassay program is designed to confirm radiological controls and to detect unknown radionuclide intakes. The purpose of the special bioassay program is to verify and monitor suspected radionuclide internal exposures. The bioassay requirements for short duration projects is also addressed in this Section.

4.1 Routine Bioassay Program

The routine bioassay program is designed to detect internal radionuclide exposures resulting in occupational doses equal to or greater than 100 mrem (0.001 Sieverts) CEDE. In addition, 10 CFR 835 requires the monitoring of internal radionuclide exposures to pregnant, minors, and members of the public exceeding 50 mrem (0.0005 Sieverts) CEDE. Individuals who have the potential to exceed these dose levels will become active participants in the routine bioassay program. The SNL/NM Internal Dosimetry Program routine bioassay program consists of baseline, routine, and termination bioassays.

4.1.1 Baseline Bioassays

The purpose of baseline bioassay is to establish the individual's internal radionuclide exposure status prior to any potential exposure at SNL/NM. Many of the radiological workers at SNL/NM have been previously employed at other facilities involved with production, processing, and storage of radioactive materials. The knowledge of all previous exposures is essential for correct assessment of dose in the SNL/NM Internal Dosimetry Program. In addition, quantifying the chronic intake of naturally occurring radionuclides (e.g., members of the uranium and thorium decay series) is needed to discriminate between occupational and background exposures. Therefore, all workers who may contact radioactive materials at SNL/NM, including sealed sources, will provide a baseline bioassay sample prior to any possible occupational exposures. Workers who meet this requirement also are required to receive Radiological Worker I training from the SNL/NM Radiation Protection Engineering Department (7713). Therefore, all workers required to complete Radiological Worker I training will also be required to provide a baseline bioassay. Visitors (i.e., minors and members of the public) who may potentially contact radioactive materials at SNL/NM will also be required to provide a baseline bioassay sample.

All baseline bioassays at SNL/NM will include urinalysis. Urinalysis provides sufficient sensitivity to detect prior exposures to most radionuclides. Workers who may contact radionuclides which can be reliably detected by WBC (e.g., radionuclides with < 250 KeV gamma emissions) will also submit to *in-vivo* bioassay techniques such as WBC.

The frequency of baseline bioassay will vary depending on the individual's employment responsibilities. Workers who only contact radioactive materials at SNL/NM will need only one baseline bioassay measurement. Other workers may visit outside facilities which produce,

process, and/or store radioactive materials for extended periods of time. These individuals will be required to submit another baseline bioassay upon returning to SNL/NM if they were potentially exposed to radioactive materials at these facilities. Baseline bioassays may also be required prior to short duration projects (e.g., environmental restoration projects, etc.) involving high hazards or to radionuclides detected in the environment.

4.1.2 Routine Bioassay

Routine bioassay is performed to meet the 10 CFR 835 dose reporting requirements and to assess the effectiveness of radioactive controls at SNL/NM. Individuals who may potentially exceed the reporting limits will be required to participate in the routine bioassay program (e.g., radiological workers and pregnant radiological workers). It is assumed that members of the public and minors would not encounter situations requiring routine bioassays. Routine bioassay frequencies are developed to ensure that any exposures resulting in doses exceeding the reporting requirements are detected. Routine bioassay at SNL/NM consists of urinalysis and, when appropriate, WBC.

Participation Requirements

Radioactive controls (e.g., air monitoring, surface contamination monitoring, SOPs, etc.) are in place at SNL/NM to insure that engineering measures are providing adequate protection against radioactive material exposures. However, these controls may not provide sufficient warning of containment failures in certain situations and/or may be compromised by human error. Because radionuclide containment designs are not infallible, routine bioassay will be performed on radiological workers who may potentially receive exposures exceeding the reporting limits. The SNL/NM Internal Dosimetry Program considers inhalation exposures to have the highest probability to occur undetected. Therefore, routine bioassay participation will be determined from the facility's potential inhalation hazard.

The inhalation hazard can be estimated by determining the facilities hazard index (HI) described in NUREG-1400 *Air Sampling in the Workplace* (NRC 91). The HI principle is based on the Brodsky's conclusion that a reasonably conservative generic estimate of the maximum fractional amount of plant throughput that gets into one employee via inhalation is 10^{-6} (Brodsky 80). The SNL/NM Internal Dosimetry Program adopts the HI principle which is consistent with the SNL/NM Air Sampling Program. The HI is calculated as follows (NRC 91):

$$HI = \frac{Q \bullet R \bullet F}{10^4 \bullet ALI \bullet C}$$

where: Q = Annual amount of the radionuclide present (uCi)

R = Release fraction

F = Modifying factor

ALI = Radionuclide annual limit of intake (uCi)

C = Confinement factor

Table 4-1 lists the release fractions from NUREG-1400 which are a function of the physical form of the radioactive material (NRC 91):

Physical Form of the Radioactive Material	Release Fraction (R)
Gases or Volatile Material	1.0
Non-volatile Powders (beta - gamma emitters)	0.01
Non-volatile Powders (alpha emitters)	0.001
Solid	0.00001
Liquids	0.0001
Encapsulated Materials	0

Table 4-1 Release Fractions from NUREG-1400

The confinement factor depends on the engineering confinement employed. Table 4-2 contains the confinement factors from NUREG 1400 (NRC 91):

Type of Confinement	Confinement Factor (C)
Glove Box	10
Hood, well ventilated	1
Open Bench, normal ventilation	0.1
Non-routine or special jobs when ventilation is unknown	0.01

Table 4-2 Confinement Fractions from NUREG-1400

The modifying factor adjusts the HI for a number of factors including the ease of dispersment and the duration that the material is present or processed. A modifying factor of 10 is used whenever energy is added to the material by process (e.g., grinding, boiling, mixing, exothermic reactions, etc.) (NRC 91).

HI for facilities containing multiple radionuclides can be estimated summing the HI for each radionuclide. A HI of 2 and 1 corresponds to potential exposures of 100 mrem and 50 mrem, respectively. Radiological workers in facilities possessing HI exceeding 2 are required to participate in the routine bioassay program. Pregnant radiological workers at facilities with HI greater than 1 will also participate in the routine bioassay program.

The HI is estimated for most facilities in the SNL/NM Air Sampling Program. In lieu of these values, routine bioassay participation can be determined by re-arranging the HI equation to solve for the quantity of radionuclides needed to achieve a HI of 2 and 1. Table 4-3 contains these quantities for the primary SNL/NM COPC assuming conservative values for R, C, and F.

Isotope	ALI (Bq)	R	C	F	Q (HI=2)	Q (HI= 1)
Tritium ¹	3 x10 ⁹	1.0	0.01	10	1.6 Ci	0.8 Ci
<u>Uranium</u> ²						
U-Nat	2 x10 ³	0.001	0.01	10	1.1 mCi	0.55 mCi
DU	2 x10 ³	0.001	0.01	10	1.1 mCi	0.55 mCi
EU	2 x10 ³	0.001	0.01	10	1.1 mCi	0.55 mCi
<u>Plutonium</u> ³						
Pu-238	2 x10 ²	0.001	0.01	10	0.11 uCi	0.055 uCi
Pu-239	2 x10 ²	0.001	0.01	10	0.11 uCi	0.055 uCi
Am-241	2 x10 ²	0.001	0.01	10	0.11 uCi	0.055 uCi

1) Assumed to be in the form of tritiated water (HTO)

2) Based on Isotopic Abundances in Table 3-X. ALI values for Class Y.

3) ALI values for Class W

Table 4-3 Quantities of SNL/NM COPC Requiring Routine Bioassay Participation

The Table 4-3 values represent the maximum radionuclide quantity exempt from routine bioassay participation. The HI calculation is considered conservative since air concentrations are often substantially lower than the predicted values. This conservatism provides a safety margin to ensure that workers are monitored in the event of a radiological incident (e.g., containment failure, accident, etc.). Therefore, the HI approach is appropriate for determining routine bioassay participation in the SNL/NM Internal Dosimetry Program.

Routine Bioassay Frequencies

The frequency of routine bioassay should be sufficient to detect reportable internal exposures which would not be detected by other means (e.g., air sampling, contamination monitoring, etc.). This frequency can be determined using the "missed dose" concept. The "missed dose" is defined as the dose which would not be detected at a given measurement interval (EGG 88). An acute intake is assumed to occur immediately following the last bioassay measurement. The magnitude of this intake corresponds to an undetected measurement during the next bioassay. Bioassay frequencies can then be developed which would limit the "missed dose" to the reporting levels (e.g., 100 mrem CEDE for radiological workers). It should be emphasized that the bioassay frequencies developed using this technique represent calendar days, not working days.

Bioassay frequencies based on the "missed dose" concept may exceed one year for some radionuclides (e.g., Class W uranium). However, such frequencies would not be sufficient to

demonstrate the annual dose reporting requirements in the 10 CFR 835 and the RCM. Therefore, the maximum bioassay frequency of workers participating in the routine bioassay program will be one year.

The benefit gained from routine bioassays in terms of a quality control measure of radiological controls must offset the costs associated with analyses, loss of work time, and worker inconvenience. For some radionuclides, the routine bioassay frequency based on "missed dose" principles are not technologically or reasonably achievable (e.g. plutonium and uranium compounds) via this criteria. The SNL/NM Internal Dosimetry Program considers 14 days to be the shortest reasonable routine bioassay frequency. This frequency is consistent with the maximum bioassay frequency for uranium exposures recommended by the Nuclear Regulatory Commission (NRC 78).

The adoption of a minimum 14 day frequency represents a routine bioassay program shortfall (i.e., unable to detect a 100 mrem "missed dose") for some radionuclide COPC. Additional emphasis will be placed on air sampling for radiological workers potentially exposed to these radionuclides (e.g., all plutonium compounds, Class Y uranium, etc.).

The calculated routine bioassay frequency will be rounded down to a frequency which facilitates program administrative concerns. The standard routine bioassay frequencies in the SNL/NM Internal Dosimetry Program are:

Bi-Weekly	14 days
Monthly	30 days
Bi-Monthly	60 days
Quarterly	90 days
Semi-Annual	180 days
Annual	365 days

The following sections describes the "missed dose" process for determining the routine bioassay frequencies for tritium, uranium, and plutonium compounds.

Routine Bioassay Frequency for Tritium

Tritium bioassay is typically performed by urinalysis. HTO is assumed to be equally distributed in body water and totally eliminated in the urine (i.e., $f_U = 1.0$). The excretion rate of HTO in urine can be determined by differentiating the HTO IRF as follows:

$$E(t)_{HTO} - \frac{C_U(t)}{C_U(0)} = -f_U \cdot \frac{d}{dt} IRF(t)_{HTO}$$

where: $C_U(t)$ = HTO urine concentration at time t (Ci/ml)

$C_U(0)$ = Initial HTO urine concentration after acute intake

f_U = The fraction of HTO body burden secreted in urine

IRF(t)_{HTO} = The HTO intake retention function
 E(t)_{HTO} = HTO urine elimination function

The HTO excretion rate equation can be arranged to yield the routine bioassay time interval corresponding to the "missed dose" as follows:

$$t = \frac{\ln\left(\frac{C_U(t)}{C_U(0) \cdot f_U \cdot 0.0695}\right)}{-0.0695}$$

The ALI for HTO is 81.2 mCi (3×10^9 Bq) (ICRP 79). The corresponding intake for 100 mrem is 2 percent of the ALI or 1.62 mCi (6×10^7 Bq). The total water content of the Standard Man is 42,000 ml (ICRP 75). Therefore, the initial urine concentration (i.e., $C_U(0)$) corresponding to a 100 mrem CEDE dose is 38.6 nCi/ml (1.4×10^3 Bq/ml). The minimum acceptable lower limit of detection (LLD) for HTO bioassay defined in the American National Standards Institute (ANSI) report N13.30 *Performance Criteria for Radiobioassay* is 1.0×10^{-2} nCi/ml (ANSI 89). This corresponds to a "missed" urine sample concentration (i.e., $C_U(t)$) of 1.0×10^{-2} nCi/ml. Inserting these values yields a 80 days routine bioassay frequency for radiological workers potentially exposed to HTO. Similarly, the routine bioassay frequency for pregnant radiological workers (i.e., 50 mrem CEDE $\approx C_U(0) = 19.3$ nCi/ml) is 70 days. Therefore, all radiological workers at tritium facilities will be monitored on a bi-monthly (i.e., 60 days) frequency.

Routine Bioassay Frequency for Uranium and Plutonium Compounds

Routine bioassay frequencies for uranium and plutonium exposures based on the "missed dose" concept can be estimated using the *INDOS* computer program. The systemic IRF for uranium (i.e., summation of organ-specific IRF) is entered to determine the uranium routine bioassay frequency, while the plutonium urinalysis pseudo IRF is used to determine the plutonium routine bioassay frequency. Table 4-4 contains the routine bioassay frequencies determined by the *INDOS* program based on urinalysis. These results represent the frequency equivalent to a mean "missed dose" of 100 mrem and 50 mrem (i.e., one half the LLD). The calculations for U-Nat, DU, and EU were based on exposures to ²³⁴U since this isotope represents the most significant internal hazard (i.e., lowest ALI).

Radionuclide	ALI (Bq)	LLD (pCi/ml)	100 mrem "missed dose" Frequency (d)	50 mrem "missed dose" Frequency (d)
<u>Uranium-234</u>				
Class D	50,000	0.1	14 (BW)	5 (BW)
Class W	30,000		2 (BW)	<1 (BW)
Class Y	1,000		<1 (BW)	<1 (BW)
<u>Plutonium-238</u>				
Class W	200	0.06	< 1 (BW)	< 1 (BW)
Class Y	500		< 1 (BW)	< 1 (BW)
<u>Plutonium-239</u>				
Class W	200	0.06	< 1 (BW)	< 1 (BW)
Class Y	600		<1 (BW)	< 1 (BW)
<u>Americium-241</u>				
Class W	200	0.06	< 1 (BW)	< 1 (BW)

BW = Bi-Weekly frequency = 14 days

Qu = Quarterly frequency = 90 days

SA = Semi-Annual = 180 days

An = Annual = 365 days

Table 4-4 Routine Bioassay Frequencies for Urinalysis based on "missed dose"

The routine bioassay frequency based on the "missed dose" concept would be less than 14 days for all uranium and plutonium compounds. The 10 CFR 835 allows the use of alternate methods for assessing internal exposures when bioassay program "shortfalls" exist. Therefore, air sampling will have a prominent role in the internal dosimetry assessment of uranium and plutonium compounds in the SNL/NM Internal Dosimetry Program.

The routine bioassay frequencies for *in-vivo* procedures can also be determined using the "missed dose" principle and the *INDOS* computer program. However, *in-vivo* bioassay for uranium and plutonium compounds are normally performed using lung counting techniques which are not available at SNL/NM on a routine basis. In practice, the sensitivity achieved

during *in-vivo* bioassay is not sufficient to detect reportable quantities of uranium and plutonium exposures. Therefore, the SNL/NM Internal Dosimetry Program will rely on urinalysis and air sampling to meet the detection requirements promulgated in the 10 CFR 835 for plutonium and uranium compounds.

Routine Bioassay Frequencies for other SNL/NM COPC

Bioassay frequencies for other SNL/NM radiologic COPC can also be calculated using the "missed dose" principle. Appendix B contains the routine bioassay frequencies for urinalysis and WBC, whenever applicable, for radionuclides which may be encountered at SNL/NM facilities.

4.1.3 Routine Bioassay Requirements for Large Facilities

The NCRP provides guidance on routine bioassay participation at facilities containing a large radiological worker population subject to the same potential exposure conditions (NCRP 87). Only a portion of the radiological workers are required to provide routine bioassays in this situation. The applicability of this approach is dependent on the time-weighted monthly average air concentration (TWA) calculated as follows (NCRP 87):

$$TWA = \frac{\sum C_i \bullet t_i}{\sum t_i} \bullet \left(\frac{100}{DAC} \right)$$

where: C_i = Air concentration radionuclide (uCi/ml)

t_i = Air monitoring time interval (h)

DAC = Derived air concentration radionuclide (uCi/ml)

TWA = Time-weighted average air concentration of the radionuclide

The TWA for multiple radionuclides can be summed to yield the total facility TWA. The facility demonstrates adequate radionuclide confinement when the total facility TWA does not exceed 2 and the largest hourly result does not exceed 6. The radionuclide confinement should be considered sufficiently effective as to require routine bioassay participation of only a representative sample of the potentially exposed individuals when these criterion are met (NCRP 87). The representative sample should comprise a portion of the most highly potentially exposed individuals and consist of 10 percent of the total radiological worker population. The minimum participation requirement is 10 radiological workers per facility (NCRP 87). Rotation of participants should be considered for personnel subject to similar potential exposures. This principle is adopted for use in monitoring large facilities in the SNL/NM Internal Dosimetry Program.

4.1.4 Routine Bioassay for Exposures to Multiple Radionuclides

Radiological workers may be exposed to compounds containing multiple radionuclides at several facilities at SNL/NM. The routine bioassay of these facilities is based on the radionuclides which constitute the significant internal hazard to workers. For the purposes of this TBD, a radionuclide is considered to be significant when it contributes 10 percent, or greater, to the total potential dose from the mixture. Significant radionuclides can be identified by estimating the internal exposure hazard index (IHI) for the heterogenous compound as follows:

$$IHI_i = \frac{\left(\frac{a_i}{ALI_i} \right)}{\sum_{i=1}^n \left(\frac{a_i}{ALI_i} \right)} \cdot 100$$

where: a_i = Activity fraction of the i^{th} radionuclide in the mixture with n radionuclides

ALI_i = The annual limit of intake of the i^{th} radionuclide

IHI_i = Internal exposure hazard index of the i^{th} radionuclide

Radionuclide compounds with an IHI greater than 10 are considered significant internal hazards. The most limiting "missed dose" frequency of the significant radionuclides is selected for the required routine bioassay frequency for the facility. When the composition of the mixture is unknown, the routine bioassay frequency will be based on the radionuclide compound with the lowest ALI.

4.1.5 Termination Bioassay

The 10 CFR 835 requires that each worker be provided a comprehensive dosimetry report within 90 days of separation from a DOE facility. An estimate of the body burdens of all detected radionuclides at the time of worker termination is necessary to meet this requirement. Therefore, all radiation workers who are leaving SNL/NM employment will provide a termination bioassay. The required bioassay techniques will typically be identical to the methods used in the baseline bioassay. However, additional bioassay tests (e.g., lung counting of plutonium workers) may be obtained at the discretion of the Internal Dosimetry Project Manager.

4.1.6 Routine Bioassay Program for Short Duration Projects

SNL/NM activities include short duration projects involving radionuclides (e.g., environmental restoration sites, one-time experiments, etc.). The need for routine bioassay program participation should be assessed since these projects often involve uncharacterized processes. Participation requirements will be evaluated by estimating the HI of the project. Emphasis will be placed on the modifying factor adjustment for short duration exposures which is calculated as follows:

$$F_T = \frac{2000}{T_{SD}}$$

where: T_{SD} = Estimated duration of short duration project (h)

F_T = Modifying factor for exposure duration

Short duration projects with HI exceeding 2 will require radiological worker participation in the routine bioassay program (1 for pregnant radiological workers). Radiological workers involved in a short duration project requiring routine bioassays will submit to a project baseline bioassay prior to exposures. Additional bioassays are required when the project duration exceeds the routine bioassay frequency and can be determined as follows:

$$RB_{SD} = \frac{T_{SD}}{RBF_{COPC}} - 1$$

where: T_{SD} = Estimated duration of short duration project (d)

RBF_{COPC} = Routine bioassay frequency of most limiting COPC (d)

RB_{SD} = Number of required short duration routine bioassays

The value for RB_{SD} is rounded to the nearest whole number. The required short duration routine bioassay frequency is then determined as follows:

$$RBF_{SD} = \frac{T_{SD}}{RB_{SD} + 1}$$

where: T_{SD} = Estimated duration of short duration project (d)

RB_{SD} = Number of required short duration routine bioassays

RBF_{SD} = Short duration routine bioassay frequency (d)

A project termination bioassay is required at the conclusion of the short duration project.

4.2 Special Bioassay Program

Special bioassays are performed on radiological workers when a radionuclide intake is suspected. The SNL/NM Internal Dosimetry Program special bioassay programs consists of radiological incident, confirmation, and exposure assessment bioassays.

4.2.1 Radiological Incident Bioassays

Radiological incident bioassays are performed whenever a reportable internal radionuclide exposure is suspected. Events indicating the occurrence of a radiological incident include the following:

- Air sampling indicates that the worker may have been exposed to air concentrations exceeding 40 DAC-hours (i.e., exposure equivalent to 100 mrem CEDE),
- Air sampling indicates that a pregnant worker or member of the public may have been exposed to air sampling exceeding 20 DAC-hours,
- A CAM alarm has occurred which was determined not to be a false alarm,
- The radiological containment at a facility with a HI exceeding 2 (1 for pregnant workers and members of the public) was determined to be compromised (e.g., unusually high air sampling and/or surface contamination monitoring results),
- The radiological worker's nasal smear sample is positive, or skin contamination is noted near the worker's nose and/or mouth,
- A radiological worker is wounded within a radiological control area, or is potentially exposed to radionuclides through an existing wound,
- Personal protective equipment (PPE) was compromised (e.g., respirator "break through", clothing tear, etc.) while the radiological worker was within a Surface Contamination Area or Airborne Contamination Area,
- A radiological worker fails to don PPE (e.g., respirators, gloves, Tyvex, etc.) prior to entering a Surface Contamination Area or Airborne Contamination Area,
- and, A radiological worker fails to follow proper protective measures within a Radiological Control Area (e.g., consumes food, smokes, etc.).

In addition, any worker who believes that he has been internally exposed to a radionuclide may request an incident bioassay. The incident bioassay will be performed with the approval of the Internal Dosimetry Project Manager.

All individuals who are potentially exposed to a radiological incident are required to submit an incident bioassay sample. At a minimum, radiological incident bioassays will consist of a 24 hour urine sample. Additional *in-vivo* and *in-vitro* analyses will be performed depending on the severity of the potential exposure. Table 4-5 provides guidance on the appropriate response to a radiological incident involving a radiological worker.

Initiating Event	<i>In-Vitro</i> Bioassay		<i>In-Vivo</i> Bioassay		
	Urinalysis	Fecal Analysis	Whole-Body Counting	Lung Counting	Wound Counting
Elevated Air Samples	> 40 DAC-h	> 40 DAC-h (Plutonium and Class Y Uranium)	> 40 DAC-h (High Energy Gamma Emitters)	> 40 DAC-h (Plutonium and Class Y Uranium)	N/A
Containment Failure	HI > 2	Optional (Plutonium and Class Y Uranium)	HI > 2 (High Energy Gamma Emitters)	Optional (Plutonium and Class Y Uranium)	N/A
Wound Exposure	All Events	Optional (Plutonium Exposures)	High Energy Gamma Emitters	N/A	All Events
Inadequate or Compromised PPE	All Events	Optional (Plutonium and Class Y uranium)	Optional (High Energy Gamma Emitters)	Optional (Plutonium and Class Y Uranium)	N/A
Positive CAM Alarm	All Events	Optional (Plutonium and Class Y Uranium)	High Energy Gamma Emitters	Optional (Plutonium and Class Y Uranium)	N/A
Potential Ingestion Exposure	All Events	Optional	High Energy Gamma Emitters	N/A	N/A

Table 4-5 Bioassay Response to Radiological Incidents

Optional procedures are subject to the professional discretion of the Internal Dosimetry Program Manager.

4.2.2 Confirmation Bioassay

All individuals who yield a positive bioassay (i.e., finding greater than the LLD, Section 5.1) will be subject to confirmation bioassays. A confirmation bioassay typically constitutes repeating the bioassay yielding the positive result. However, additional bioassay procedures can be performed (e.g., *in-vivo* bioassay procedure) at the Internal Dosimetry Program Manager's discretion. The monitored worker may be required to participate in the exposure assessment bioassay program (Section 4.5) if the subsequent confirmation bioassay validates the positive finding. Another confirmation bioassay is required if the initial finding is not validated. The initial finding is considered a false positive when two successive confirmation bioassays are negative (i.e., finding below the LLD).

4.2.3 Exposure Assessment Bioassay

All radiological workers yielding a confirmed positive bioassay which corresponds to a reportable exposure (i.e., 100 mrem CEDE) will participate in the exposure assessment program. The required number and frequency of exposure assessment bioassays is dependent on several factors including the following:

- The potential magnitude of the intake,
- Biokinetics of the radionuclide compound,
- Physical half-life of the radionuclide,
- Analytical sensitivities for the radionuclide compound,
- Health of the monitored worker,
- and, The number of measurements required for an accurate dose assessment.

At a minimum, one exposure assessment bioassay representing each biokinetic model component should be collected. For example, the IRF for cobalt compounds consist of a 4 component model expressed as follows:

$$R(t)_{Co} = 0.5 e^{\left(\frac{-0.693 \bullet t}{0.5}\right)} + 0.3 e^{\left(\frac{-0.693 \bullet t}{6}\right)} + 0.1 e^{\left(\frac{-0.693 \bullet t}{60}\right)} + 0.1 e^{\left(\frac{-0.693 \bullet t}{800}\right)}$$

If possible, at least one exposure assessment bioassay sample should be collected for each of the 0.5, 6, 60, and 800 day components. Individual exposure assessment durations and bioassay frequencies will be developed for each participant in the exposure assessment program to meet the dosimetry evaluation and reporting requirements in the 10 CFR 835. These requirements will be defined by the Internal Dosimetry Program Manager.

5.0 Bioassay Analysis

The goal of bioassay is to accurately estimate the initial intake of internally deposited radionuclides in the monitored worker. This Section provides guidance on bioassay sample collection and analyses in the RPID Project. Quality control and sensitivity requirements are defined to ensure that the bioassay data is sufficiently reliable to meet the objectives of this TBD.

5.1 Bioassay Sample Collection

Accurate internal dosimetry assessments rely on the collection of representative bioassay samples. The principle concern is preventing sample contamination from radionuclides not internally present within the monitored worker. All internal dosimetry program participants should receive training regarding the purpose, rationale, methods, and scheduling of bioassays to facilitate proper sample collection.

The following individual information is obtained during all bioassays to facilitate accurate documentation of analytical results:

- Full name and former names,
- Social security number or passport number and country,
- Date of birth,
- Sex,
- Employment status and job title,
- Principle facility type and building number, and
- Organization code.

Additional sample collection information may be required depending on the bioassay sample type.

5.1.1 *In-Vitro* Samples

In-vitro bioassay samples are collected in radiologically clean environments, preferably outside of SNL facilities. Samples are collected in "clean" areas when collection at SNL facilities is necessary. Participants will follow all personal decontamination guidelines (e.g., showering, change clothing, hand-washing, etc.) prior to collecting bioassay samples at SNL.

Participants should be provided an adequate supply of sample containers for the required bioassay. Single use containers are recommended to prevent cross contamination between individuals. Sample containers should be unbreakable and selected to minimize sample loss from radionuclides adhering to container walls. Sample containers can also be acidified to minimize sample precipitation effects. All biological samples are subject to deterioration by

bacterial action which may interfere with subsequent analyses (ICRP 87). Preservatives or refrigeration should be considered whenever samples are not promptly analyzed.

Urinalysis

Radionuclides that enter systemic circulation are distributed throughout the body, chemical properties may cause them to concentrate in selected organs or tissues. The radionuclide is then metabolized back into the bloodstream, usually in a chemical form that can be excreted from the body. One common excretion pathway involves transport through the kidneys, where the radionuclide is filtered out of the blood. After filtration the radionuclides are transported through the normal urine pathway to the bladder and excreted. The rate at which a radionuclide follows this pathway depends upon many chemical and biological factors.

Over a 24 hour period the influence of various biological and chemical factors on the radionuclide excretion rate tend to be less significant. To lessen the impact of these normal variations urine samples are usually collected over a 24-hour period. Using 24-hour samples gives a more accurate result than single-void or “spot” samples. Following a suspected intake 24-hour samples are collected from the exposed workers. If a 24-hour sample is not practical, then corrections can be made as described in this section. Routine bioassay samples are not collected for a fixed time. The participating workers are requested to collect urine until the 1500 ml bioassay kits are full. This ensures that the analysis laboratory has a large enough sample volume.

24-hour samples are not required for some radionuclides (e.g., tritium). Analysis of a single voiding is sometimes sufficient to give adequate evidence of exposure or lack thereof. Consideration of sample collection time should be given when collecting single voiding samples. Samples collection at the end of the work week would be more sensitive for detecting workplace exposures, while samples collected after a long period of no exposure (e.g., vacation, change of work assignment, etc.) would be more sensitive to detect exposures from environmental radionuclides. In addition, samples collected directly after exposure are influenced by the initial rapid clearance fraction. This time period is often poorly represented by ICRP models and may lead to intake over-estimations. The Nuclear Regulatory Commission recommends that specimen collection be delayed between 48 and 96 hours after the most recent exposure opportunity for uranium urinalysis (NRC 78).

Urine samples which are not true 24-hour samples can be corrected to account for abnormal conditions of high or low fluid intake, or excessive loss of water by perspiration. Corrections can be made based on the specific gravity as follows (NOSH 74):

$$C_{24-h} = C_M \cdot \frac{1.024 - 1}{SG_M - 1}$$

where: C_M = Measured urine concentration (uCi/ml)
 SG_M = Measured urine specific gravity

C_{24-h} = Estimated 24-hour urine concentration (uCi/ml)

Another technique is based on creatinine measurements as follows (Jackson 66):

$$C_{24-h} = C_M \cdot \frac{CR_E}{CR_M}$$

where: C_M = Measured urine concentration (uCi/ml)

CR_E = Expected creatinine daily content (1.7 g for men, 1.0 g for women)

CR_M = Measure urine creatinine content (g)

C_{24-h} = Estimated 24-hour urine concentration (uCi/ml)

Sample volumes of actual 24-hour samples may be indicative of a need for similar corrections (e.g., urine volume too small to represent a true 24-hour sample). Urinalysis in the RPID Project will involve true 24-hour collections when single void samples are considered inappropriate. Routine 24-hour samples may be repeated if they are considered non-representative. Adjustments for the hydration status of the monitored individual are considered when the samples are being used to assess internal exposures.

In the event of a radiological incident, the immediate single-void urine sample should not be included within the 24-hour sample. This sample is often representative of pre-accident conditions and may dilute the 24-hour sample assessment. The initial single-void sample may be used to represent pre-accident baseline information.

Fecal Analysis

Initial fecal samples may not indicate positive results since transmission times through the GI tract are often variable and can be substantially delayed. The uncertainty associated with this variability is magnified by the ICRP GI tract model which predicts significant radionuclide compound clearance by an early fecal excretion mechanism. Therefore, collecting all fecal excretions over the first five or six days following a potential serious exposure is recommended to reduce the inherent uncertainties (Skrable 87). Fecal samples may be assayed daily to provide estimates of the severity of the exposure event. However, the most accurate intake estimate can be made by mathematically combining all of the daily fecal analyses into one composite sample on a weight basis.

The accuracy of dose estimates from fecal sample results is increased after several days post intake. This allows for the early clearance through the GI tract of ingested material or material translocated out of the lung directly into the GI tract. Radionuclides in fecal mater several days after the event represent the fraction of the intake that was metabolized through the systemic whole body.

Nasal Smears

The representativeness of nasal smears are greatly enhanced when they are collected promptly after suspected intakes and before the individual blows his nose or showers (NCRP 80). The sample can be collected on a moist cotton-tipped applicator or on filter paper on a swabstick. Separate smears are taken from each nostril. Nasal smears should be dried prior to any alpha analytical measurements. Great care should be taken to prevent hand and face contamination from compromising the sample. Since the activity on nasal smears represents activity removed from the body, use of nasal smear data to quantify an intake is questionable. Result of nasal smear analysis may be used as an indicator of intake, which is quantified by other bioassay techniques.

5.1.2 *In-Vivo* Measurements

Counting facilities are designed to minimize the effects of background radiation during the bioassay. All materials used during an *in-vivo* bioassay should be selected for low radionuclide content. Foot traffic through the bioassay area should be minimal to prevent introducing effects from environmental radionuclides.

Surface contamination can severely distort *in-vivo* bioassay measurements since photon attenuation is minimal. Therefore, a whole body frisk is performed prior to measurement. Workers found to be contaminated are referred to Radiation Protection Operations personnel for decontamination. Disposable clothing should be provided to prevent potential interference from radionuclides on the worker's clothing.

5.2 Intake Identification and Analytical Sensitivity Requirements

Identifying radionuclide intakes originating from SNL facilities requires the consideration of radiation detection statistics and background radionuclide interference. In addition, the analytical method used must be sufficiently sensitive to detect intakes resulting in reportable exposures (i.e., 100 mRem CEDE). The following sections outline the intake identification process and sensitivity requirements for the RPID Project. Figure 5-1 summarizes the bioassay interpretation process for the RPID Project.

5.2.2 Influence of Measurement Uncertainty

The uncertainty associated with the measurement technique may exceed the measured value in some situations (e.g., low count analytical results). A more precise analytical method should be considered to clarify the finding. In the absence of a more precise technique, positive findings which are less than the associated measurement uncertainty are considered to indicate a negative radionuclide intake finding in the RPID Project.

5.2.3 Influence of Non-Occupational Intakes

Positive bioassays are expected for radionuclides which can be obtained from environmental sources outside of SNL facilities (e.g., dietary uranium intakes,). All positive results are compared to the expected distribution of background levels obtained from an un-exposed population.

An initial study has been performed to establish the natural background level for the total uranium content in urinalysis. The average result for 14 people was 0.16 ug/L with a standard deviation of 0.05 ug/L. These results correspond to an investigation level of 0.31 ug/L (mean + 3σ) for total uranium measurements in urine samples. Total uranium values exceeding 0.31 ug/L indicate potential occupational exposures to uranium compounds. The findings of this study are expected to be revised once a more comprehensive internal radionuclide background study at SNL is completed.

For other radionuclides found in nature, (cesium from bomb tests, iodine from nuclear medicine, etc.) RPID personnel may discuss the bioassay results with the worker to determine if the cause is occupational exposure. The investigation level for radionuclides not measured in an un-exposed population is zero. Ninety nine percent of all background measurements are expected to fall below the investigation level.

5.2.5 Confirmation Measurements for Positive Bioassay Findings

When appropriate, a confirmation measurement may be performed to validate positive measurements. If possible a different portion (i.e., "split" sample) of the original bioassay sample may be re-analyzed and compared to the original result. The initial measurement is considered validated if the re-analysis is also positive. A third portion may be analyzed to validate the positive finding when necessary. The initial finding is considered a false positive when the two successive measurements are both negative.

5.2.5 Bioassay Sensitivity Requirements

Bioassay sensitivity requirements are established by optimizing the requirements of two conflicting goals. The sensitivity must be low enough to minimize the number of false negatives (i.e. missed dose), without adding the administrative burden of a high number of false positives. For most radionuclides this is accomplished by using the sensitivity requirements specified in ANSI N13.30 *Performance Criteria for Radiobioassay*. Minimum detectable activities (MDA) are defined in the ANSI to be equivalent to a measurement response corresponding to a 5 percent chance of a false positive finding (i.e., Type I error) and a 5 percent chance of a false negative finding (i.e., Type II error). The MDA is calculated as follows (ANSI 89):

$$MDA = \frac{(4.65 \cdot \sigma_B + 3)}{K \cdot V \cdot T}$$

where: σ_B = standard deviation of a blank response

K = analytical system counting efficiency (cpm/dpm)

V = sample volume or mass quantity

T = counting time of analyzed sample

MDA = minimum detectable activity (dpm/mass or volume quantity)

The acceptable MDA is achieved by selecting an appropriate combination of analytical equipment and facilities, sample processing, and counting techniques. The ANSI N13.30 standards for *in-vitro* bioassays are provided for ten radionuclide classes depending on the principle mode of radiation emission. Table 5-1 provides the acceptable MDA for bioassay by urinalysis, while Table 5-2 lists fecal analysis acceptable MDAs (ANSI 89).

ANSI Category	Radionuclide	Acceptable MDA (pCi/L) for Urinalysis
I. Average Beta < 100 KeV	Tritium (3H)	10,000
	Carbon-14	10,000
	Sulfur-35	100
	Promethium-147	10
	Lead-210	5
	Radium-228	5
	Plutonium-241	5
II. Average Beta > 100 KeV	Phosphorus-32	20
	Strontium-89/90	20
	Iodine-131	100
III. Alpha	Polonium-210	0.1
	Radium-226	0.1
	Thorium-228,230,232	0.1
	Uranium-234,235,238	0.1 ¹
	Neptunium-237	0.06
	Plutonium-238,239/240	0.06
	Americium-241	0.06
	Curium-242,244	0.06
IV. Mass Determination	Uranium (Natural)	5 (ug/L) ¹
V. Gamma or X-rays	Emitters > 100 KeV	50/A ²
VI. Gamma or X-rays	Emitters ≤ 100 KeV	50/A ²

¹ Either MDA is appropriate for uranium bioassay

² Abundance (A) is the photons per disintegration of the most abundant photon with energy greater than 100 KeV associated with the decay of the radionuclide of interest.

Table 5-1 Acceptable MDA for Urinalysis of Various Radionuclides

ANSI Category	Radionuclide	Acceptable MDA (pCi/Sample) for Fecal Analysis
VII. Alpha	Uranium-234,235,238	1
	Thorium-228,230,232	1
	Plutonium-238,239/240	1
	Americium-241	1
VIII. Average Beta > 100 KeV	Strontium-89/90	20
	Strontium-90	20
IX. Gamma or X-rays	Emitters > 100 KeV	50/A ¹
X. Gamma or X-rays	Emitters ≤ 100 KeV	50/A ¹

¹ Abundance (A) is the photons per disintegration of the most abundant photon with energy greater than 100 KeV associated with the decay of the radionuclide of interest.

Table 5-2 Acceptable MDA for Fecal Analysis of Various Radionuclides

The ANSI acceptable MDA standards for WBC can be derived from the following relationship:

$$AMDA_{WBC} = \frac{20}{a}$$

where: a = number of photons per transformation associated with the photon being measured.

AMAD_{WBC} = acceptable MDA for WBC (nCi)

The ANSI acceptable MDA for urinalysis, fecal analysis, and WBC shall be used in the RPID Project. All bioassays performed for the RPID Project will, at a minimum, meet the ANSI acceptable MDA standards.

5.3 In-Vitro Bioassay Analytical Techniques

Selecting the appropriate analytical method is dependent on the radiation emitted from the COPC and on the degree of specificity required (i.e., gross versus isotope-specific analyses). Numerous analytical techniques are available to analyze *in-vitro* bioassay samples. The primary objective of each analytical method used is meeting the ANSI sensitivity standards. The following should be considered when selecting the appropriate analytical technique:

- Degree of analytical sensitivity required,
- Availability of specific analytical tests,

- Sampling requirements associated with specific analytical tests (e.g., specialized equipment, costs, etc.),
- Time constraints for analytical results (i.e., turn-around time), and
- Costs associated with sample analysis.

The SNL Radiation Protection Sample Diagnostics(RPSD) Project maintains automated gross and qualitative isotopic radiation measurement capabilities. Because the provided analytical services are less expensive and more convenient than contract laboratory services, for some radionuclides, use of RPSD capabilities is emphasized in the RPID Project. However, RPSD does not have chemical separation capabilities (i.e., required for isotopic separations) or fecal analysis expertise at this time. A contract laboratory capable of meeting the required MDA's is used for isotopic analyses and fecal analysis of *in-vitro* bioassay samples in the RPID Project.

5.3.1 Tritium *In-Vitro* Bioassay Measurement

Tritium urinalysis is almost exclusively performed using liquid scintillation counting (LSC). RPSD maintains a LSC system which meets the required standards for tritium sensitivity. Typical turn-around times for tritium urinalysis achieved by RPSD is sufficient for both routine and special bioassay applications. Therefore, Department 7715 is expected to be the primary provider of tritium urinalyses for the RPID Project.

5.3.2 Transuranic *In-Vitro* Bioassay Measurements

Transuranic *in-vitro* bioassays may be analyzed using qualitative and quantitative techniques. RPSD maintains a qualitative measurement capacity which is suitable for analyzing routine bioassay samples. Samples requiring isotopic information must be analyzed using alpha spectroscopy. RPSD does not maintain the required equipment to support this analysis. Therefore samples requiring this type of analysis are sent to a contract laboratory

The turn-around time of contract isotopic analyses may exceed 30 days. Decisions regarding confirmation and exposure assessment bioassays may have to be made prior to receiving the isotopic information. Therefore, portions of all special bioassays will be analyzed using RPSD equipment when practicable. Preliminary decisions regarding intake confirmation and dosimetry are based on conservative assumptions using survey. The dosimetry decisions/actions are revised once the isotopic information is obtained. Because of its dose management importance, all isotopic-specific analyses for special bioassays are performed on a "rush" basis.

5.3.3 In-Vitro Bioassay Measurements for Fission / Activation Products

Bioassay sample measurement for fission / activation products varies depending on their respective radiation emission characteristics. RPSD facilities include gamma spectroscopy and are used to expedite sample analysis time. A contract laboratory is used to quantify special bioassays when necessary (e.g., isotopes of thorium, strontium-89/90). Appendix B contains the suggested analytical techniques for the SNL COPC.

5.4 In-Vivo Bioassay Analysis

The RPSD WBC system is a Canberra ACCUSCAN II. This system has two HPGe detectors which scan the worker as they stand in a shielded booth. Gamma spectroscopy data is analyzed using a DEC AXP computer, which runs Canberra's ABACOS-+ software. This software is an industry standard and includes an algorithm which compensates for compton scattering effects from extraneous background radiation sources (e.g., radionuclides within the subject and/or building materials, cosmic rays, etc.) using internal spectrum stripping techniques. The net output spectrum is then compared against a library of gamma spectrum responses for various isotopes of interest. Positive findings are quantified using calibration factors obtained from a tissue equivalent phantom. Calibration geometry's are maintained for whole body, lung, GI tract, and Thyroid. Calibration is performed using the RMC II phantom, which yields results in good agreement with the more complex Livermore phantom. A mixed gamma source with emissions from 88 to 1836 keV is used. During routine bioassay the whole body geometry is used. It should be noted that the accuracy of the WBC measurement decreases as the difference between measurement and calibration geometry's increases (e.g., individual is substantially larger or smaller than phantom, etc.).

The RPSD WBC system meets the ANSI acceptable MDA requirements for most radionuclides of interest which emit photons with energies greater than 250 KeV. RPSD staff participate in industry wide studies of WBC sensitivity ensuring high performance standards are maintained. Radionuclides which emit less energetic photons (e.g., isotopes of plutonium, etc.) may be assessed using *in-vivo* techniques at a contract laboratory (i.e., lung counting at Los Alamos National Laboratories) during special bioassay applications.

5.5 Bioassay Quality Control Measures

Quality Control (QC) measures are required to ensure that the reported analytical results accurately reflect the conditions of the bioassay sample. Artificial influences due to sample collection, preparation, and analysis indiscretions must be identified to prevent error propagation into the subsequent exposure assessment phase.

The Statement of Work for the contract laboratory includes extensive QC requirements. These requirements include committing to ANSI N13.30 Draft-Performance Criteria (Revision May 6, 1994). Additional requirements on the use of QC charts for counting instruments are also included. Failure to meet the QC does not necessarily invalidate the analytical result. However,

QC transgressions should be considered when evaluating marginal positive results. Contradictory analytical findings with marginal QC performance should be considered for re-assessment. The following sections summarize the principle QC measures in the RPID Project.

5.5.1 Sample holding times and preservation requirements

In-vitro Bioassay are refrigerated to minimize biological degradation. Shipments to offsite facilities are made in coolers with refrigerant materials. Upon receipt the contract lab is required to immediately refrigerate the samples. In addition all sample containers are rinsed with nitric acid to inhibit bacterial growth. The analytical result is given a "J" (i.e., estimated) qualifier when any QC criteria are not met. The holding time and preservative requirements for fecal samples are not well defined. In the absence of such standards, urinalysis requirements may be applied to fecal samples. However, reviewer discretion should be applied as required.

5.5.2 Analytical System Calibration

Both *in-vivo* and *in-vitro* analytical systems must be calibrated to ensure that the instrument is capable of producing acceptable quantitative data. Calibration of all analytical equipment should be performed at least annually, or whenever performance checks indicate an unacceptable change in system performance. All calibrations should be performed using standard instrument calibration sources. Performance checks should be performed daily with standard calibration sources to confirm that the equipment response is reliable. Analytical results are given a "J" qualifier if these measures are not followed.

5.5.3 Blank sample responses

Blank analyses indicate the instrument's response to the background radiation environment. In addition, the presence and magnitude of analytical equipment contamination can be established by performing blank analyses. Blank samples should be analyzed concurrently with bioassay samples with a 5 percent frequency (i.e., one blank per twenty bioassay samples). The blank matrix should be similar to the bioassay being analyzed (e.g., artificial urine would be suitable for urinalysis). Acceptable responses to blank samples should be established based on instrument-specific trend analyses. Analytical results will, at a minimum, be given a "J" qualifier whenever a blank measurement deviates from the normal response.

5.5.4 Chemical recovery responses

Known quantities of radionuclides or other tracers are sometimes introduced into the *in-vitro* bioassay sample to evaluate the chemical recovery of the analytical technique. Comparing the known and measured values provides a measure of the influence of chemical separations and intermittent handling procedures on the final result. In addition, these results also indicate the influence of the bioassay sample matrix on the final result. The chemical recovery for each analytical method can be calculated as follows:

$$CR = \frac{R_M}{R_S} \bullet 100$$

where: R_M = measured response (pCi/volume or mass unit)

R_S = known "spiked" quantity (pCi/volume or mass unit)

CR = percent chemical recovery of the analytical procedure

Table 5-4 provides a range of data qualifiers depending on the chemical recovery of the analytical procedure.

Percent Chemical Recovery Range	Data Qualifier
50 to 105	Acceptable for Use (none)
> 105	Estimated (J)
20 to 50	Estimated (J)
< 20	Unusable (R)

Table 5-4 Data Qualifiers for Various Chemical Recoveries

The large acceptable chemical recovery range reflects the allowable variability for sample matrix effects. Chemical recoveries less than 20 indicate unacceptable matrix effects levels. Another analytical technique should be considered in these events.

5.5.5 Laboratory control sample responses

Laboratory control samples (LCS) provide a measure of the accuracy of the analytical technique.

The LCS contains a known radionuclide quantity within a sample matrix similar to the bioassay sample. The LCS is then analyzed using the same methodology of the bioassay samples.

Comparisons of the known value with the measured quantity provides a measure of the bias inherent within the entire analytical process. LCS are typically evaluated with a 5 percent frequency (i.e., one LCS per twenty bioassay samples). The LCS recovery is then calculated as follows:

$$LCS = \frac{R_M}{R_{LCS}} \bullet 100$$

where: R_M = measured response (pCi/volume or mass)

R_{LCS} = known LCS quantity (pCi/volume or mass)

LCS = percent recovery of the lab control sample

Table 5-5 provides data qualifiers for a range of LCS recovery values. Failure to meet the target percent recovery range results in the issuance of a "J" qualifier. Failure to meet the minimum recovery standards results in a "R" (i.e., unusable) qualifier.

LCS for Urinalysis		LCS for Fecal Analysis	
LCS Recovery Range	Analytical Qualifier	LCS Recovery Range	Analytical Qualifier
80 to 120	Acceptable for Use (none)	70 to 130	Acceptable for Use (none)
> 120	Estimated (J)	> 130	Estimated (J)
50 to 79	Estimated (J)	40 to 69	Estimated (J)
< 50	Unusable (R)	< 40	Unusable (R)

Table 5-5 Data Qualifiers for Various Laboratory Control Sample Recoveries

5.5.6 Analytical sensitivity requirements and data qualifier assignments

All bioassays must meet the ANSI acceptable MDA standards. A unusable "R" qualifier may be assigned if the achieved MDA significantly exceeds this requirement. The bioassay sample should be considered for re-analysis in this event if possible. Decisions regarding qualifiers or re-analyses of bioassays not meeting the ANSI requirements are made at the discretion of the Internal Dosimetry Project Leader.

Results and uncertainties should be reported for all bioassays regardless of the size or sign of the result. The reported uncertainty should include all sources of uncertainty (e.g., counting and established systematic uncertainties, etc.). The overall uncertainty should be reported at the 95% confidence level (i.e., 2 σ level). The following qualifiers are assigned based on the reported analytical results:

- J The reported result is estimated.
- R The reported result is unusable.
- B The result from a reagent blank associated with this sample was greater than or equal to the MDA.
- L A lab control sample associated with this sample had a low bias or this sample had a low tracer recovery.
- H A lab control sample associated with this sample had a bias or this sample had a high tracer recovery.

- P The sample result is preliminary.
- X Some data necessary to compute the sample result, error, or MDA was manually entered or modified.
- M There were two or more results available for this analyte. The reported result may not be the same as the raw data.

The analytical laboratory should be contacted if discrepancies are identified in assigning data qualifiers.

5.5.8 Spectroscopy identification responses

Analytical techniques which employ spectroscopy (e.g., alpha and gamma spectroscopy) require energy calibrations to ensure accurate isotopic identifications. Scattered radiation from other radionuclide emissions may also decrease the accuracy of isotope identifications via spectroscopy. The degradation's caused by these effects can be identified by observing the analytical data. Table 5-6 lists the data qualifiers for typical spectroscopy observations.

Alpha spectroscopy degradation is not tolerated since chemical separations are designed to remove interfering radionuclides. Gamma spectroscopy degradation can be detected by observing the response to a known radionuclide emission. Unidentified emissions are given an uncertain identification (UI) qualifier when gamma spectroscopy degradation is not noted.

Spectroscopy Observation	Possible Causes	Data Qualifier
<u>Alpha Spectroscopy</u> 1) Peak shift greater than 40 KeV 2) Overlapping Energy Peak Identified	1) Energy Calibration Required, Power Drift 2) Incomplete Chemical Separation	1) Unusable (R) 2) Unusable (R)
<u>Gamma Spectroscopy</u> 1) Peak shift of known radionuclide greater than 2 KeV. 2) Peak greater than 2 KeV shift from known radionuclide library 3) 50% Gamma Abundance not detected for identified radionuclide	1) Energy Calibration Required, Power Drift 2) Unidentified or Incomplete Spectrum Detected 3) Unidentified or Incomplete Spectrum Detected	1) Unusable (R) 2) Uncertain Identification (UI) 3) Uncertain Identification (UI)

Table 5-6 Data Qualifiers for Various Spectroscopy Observations

5.5.9 Analytical system performance records

The general performance of the analytical system can be assessed through the following observations:

- Response fluctuations to radiation background,
- High background responses,
- Energy calibration shifts, and
- Loss of resolution.

These findings suggest an overall decrease in analytical quality. Responsible analytical laboratory personnel monitor these parameters and correct any transgressions. The analytical reviewer may assign data qualifiers or request re-analyses when these effects are noted.

6.0 Dosimetry Assessment

The ICRP provides a methodology for measuring the detrimental effects of radiation exposure. Radiation detriment is measured in terms of dose and is categorized into two general classes:

- Stochastic effects such as carcinogenesis or genetical disease which is assumed to be a function of dose without threshold,
- and, Non-stochastic effects such as opacity of the lens which is assumed to be a function of dose with threshold.

The 10 CFR 835 and RCM requirements are designed to limit the occurrence of stochastic effects to an acceptable level and to prevent the occurrence of non-stochastic effects. The radiation detriment to radiological workers can be monitored and compared to the 10 CFR 835 and RCM requirements using the ICRP dosimetry methodology which is summarized in this Section.

6.1 ICRP Internal Dosimetry Methodology

The 10 CFR 835 dose limits are expressed in terms of TEDE which is the summation of dose from external radiation sources with the CEDE from internal radionuclide depositions. The committed dose equivalent from an internally deposited radionuclide is calculated as follows (Turner 88):

$$H_{50,T} = \sum_i Q_i \bar{D}_{50,i}$$

where: Q_i = Quality factor of the i^{th} radiation

$\bar{D}_{50,i}$ = Absorbed dose averaged over an organ or tissue from the i^{th} radiation over 50 years (rad)

$H_{50,T}$ = Committed dose equivalent to organ or tissue T (rem)

The quality factor (Q) is a function of the linear energy transfer of the emitted radiation. Table 6-1 contains the ICRP values for Q which are assumed to be constant for each radiation type (ICRP 79):

Radiation Type	Quality Factor (Q)
Beta particles, electrons, gamma rays, X-rays, and Bremsstrahlung	1
Spontaneous fission neutrons and protons	10
Alpha particles, heavy recoil particles, and fission fragments	20

Table 6-1 Quality Factors (Q) for Various Radiation Emissions

The organ of interest (i.e., target organ) can be irradiated from radionuclides contained within or from radionuclides deposited in some other organ (i.e., source organ). The committed dose equivalent equation can then be re-written as follows (ICRP 79):

$$H_{50}(T \leftarrow S)_i = Q_i \overline{D}_{50}(T \leftarrow S)_i$$

It should be noted that the both the target and source organ can be the same.

6.1.1 Committed Dose Equivalent

The average committed absorbed dose delivered to the target organ depends on the specific effective energy (SEE) per transformation and the number of transformation (U_s) in the source organ over 50 years. The total committed dose equivalent to the target organ from all radiation emissions from radionuclides contained in all source organs can be expressed as follows (ICRP 79):

$$H_{50,T} = 1.6 \times 10^{-10} \sum_s \sum_j \left[U_s \sum_i SEE(T \leftarrow S)_i \right]_j$$

where: U_s = The number of transformation of the j^{th} radionuclide

$SEE(T \leftarrow S)$ = The specific effective energy to the target organ from the i^{th} radiation emission from the j^{th} radionuclide located in the s^{th} source organ (MeV/g)

$H_{50,T}$ = Total committed dose equivalent to the target organ from all radiation emissions from all radionuclides deposited in all source organs (rem)

The SEE is the quality factor weighted energy absorbed per gram of the target organ from the j^{th} radionuclide and is expressed as follows (ICRP 79):

$$SEE(T \leftarrow S)_j = \frac{1}{M_T} \sum_i Y_i E_i AF(T \leftarrow S)_i Q_i$$

where: M_T = Mass of the target organ or tissue (g)

Y_i = Yield of the i^{th} radiation per transformation

E_i = Average energy of the i^{th} radiation (MeV)

$AF(T \leftarrow S)$ = Fraction of energy emitted from source organ absorbed in the target organ

Q_i = Quality Factor of the i^{th} radiation

$SEE(T \leftarrow S)$ = Total specific effective energy per transformation to the target organ from the j^{th} radionuclide deposited in the source organ (MeV/g)

The mass of the target organ is expected to vary widely in a worker population. Individual adjustments are considered to be impractical for most dosimetry applications. The ICRP Publication 23 *Reference Man: Anatomical, Physiological, and Metabolic Characteristics* provides anatomical data for a theoretical reference individual (ICRP 75). The "Reference Man" is defined as being between 20 and 30 years of age, weighing 70 kg, is 170 cm in height, lives in a climate with an average temperature between 10 to 20 C, is caucasian, and is a Western European or North American in habitat and custom (Turner 92). The RPID Project adopts the "Reference Man" concept for use in evaluating internal dosimetry. Adjustments from these assumptions may be made to evaluate significant internal exposures as necessary.

The absorption fraction (AF) vary widely depending on the target to source organ tissue distance and the type of radiation being emitted. Values range from 0 for non-penetrating radiations emitted in source organs distant from the target organ to 1 for non-penetrating radiations originating from the source organ which is also the target organ. The AF for penetrating radiations (e.g., gamma and x-rays) is difficult to estimate and is dependent on their energy; the size, density, and relative position of the source and target organs; and the properties of the intervening tissues (Turner 92). Fortunately, the Medical Internal Radiation Dose (MIRD) Committee of the Society of Nuclear Medicine has determined values for AF for a number of source and target organs in Reference Man using Monte Carlo computer techniques. The RPID Project will adopt the MIRD AF values whenever SEE is evaluated.

Values for radiation yields (Y_i) and energy per transformation (E_i) are dependent on the decay scheme of the evaluated radionuclide. Decay schemes used in ICRP 30 are detailed in ICRP Publication 38 *Radionuclide Transformations, Energy, and Intensity of Emissions*. The RPID Project will refer to the ICRP 38 data whenever SEE is evaluated.

The number of transformations of a radionuclide in any organ or tissue (U_S) is the time integral of radionuclide activity within the organ or tissue over the evaluated time period. The standard time period evaluated by the ICRP is 50 years. The function describing uptake and retention of radionuclides within an organ following exposure (e.g., inhalation, ingestion, etc.) may be very complex and therefore is convenient to describe by simple models which facilitate calculation

and yield sufficiently accurate results (ICRP 79). Examples of these models include the ICRP Lung and GI model and the IRF for each respective radionuclide. These models consist of a number of transfer compartments in which radionuclide compounds decay or are translocated to other compartments. Radionuclide compounds are assumed to be uniformly distributed within the compartment and any radioactive progeny are assumed to be metabolized like the parent. The *INDOS* computer program is capable of estimating U_s for various radionuclide compounds entering the body from various exposure pathways.

The estimated committed dose equivalent evaluated for each target organ can then be used to evaluate compliance with the 10 CFR 835 non-stochastic dose limits.

6.1.3 Committed Effective Dose Equivalent

The total detriment from internal radionuclide exposures is expressed as the CEDE. The CEDE is estimated by summing the weighted committed dose equivalents of organs or tissues exposed to an internal intake as follows:

$$H_{wb} = \sum_T w_T H_{50,T}$$

where: w_T = Weighting factor of tissue T

$H_{50,T}$ = Committed dose equivalent to tissue T (rem)

H_{wb} = Committed effective dose equivalent to the whole-body (rem)

The weighting factor (w_T) represents the ratio of the stochastic risk resulting from tissue T to the total risk when the whole body is irradiated uniformly. Table 6-2 provides the weighting factors adopted in ICRP 30 (ICRP 79).

Organ or Tissue	Weighting Factor (w_T)
Gonads	0.25
Breast	0.15
Red Bone Marrow	0.12
Lung	0.12
Thyroid	0.03
Bone Surfaces	0.03
Remainder	0.30

Table 6-2 ICRP 30 Tissue Weighting Factors

The remainder fraction ($W_T = 0.30$) represents W_T values of 0.06 assigned to the five remaining organs or tissues that receive the highest committed dose equivalents. This assumption ignores exposures to all other unspecified tissues. The estimated CEDE can then be compared against the 10 CFR 835 stochastic dose limits.

6.1.4 Derived Dosimetry Standards

Derived dosimetry standards are designed to meet the 10 CFR 835 dose limits and are derived using ICRP 30 methodologies. Allowable intakes of radionuclide compounds can be estimated using the following relationships:

$$I \sum_T w_T H_{50,T} \leq 5rem$$

$$I \bullet H_{50,T} \leq 50rem$$

where: I = Annual intake of the specified radionuclide by ingestion or inhalation (uCi)

w_T = Tissue weighting factor

$H_{50,T}$ = Committed dose equivalent per unit intake (rem/uCi)

The ALI is defined as the largest value for the intake (I) which satisfies both of these relationships and are exposure pathway specific. Values are published for inhalation and ingestion exposures in ICRP 30.

The derived air concentration (DAC) represents the maximum radionuclide air concentration that satisfies the allowable intake relationships. It is calculated using Reference Man assumptions for annual volume of air breathed during light activity as follows:

$$DAC = \frac{ALI_{inh}}{2400}$$

where: ALI_{inh} = Radionuclide specific annual limit of intake for the inhalation pathway (uCi)

DAC = Radionuclide specific derived air concentration based on Reference Man assumptions (uCi/m³).

Values for DAC are also published in ICRP 30. Because dose is considered directly proportional to intake, dosimetry assessment can be greatly simplified using the derived dosimetry standards.

6.2 Intake and Dose Assessment

Internally deposited radionuclide intakes can be estimated from bioassay data using appropriate metabolic models. The corresponding dose is calculated from the estimated intake using the appropriate derived dosimetry standard. The following information should be obtained to facilitate intake and dose assessments in the RPID Project (NCRP 87):

- Radionuclide species from bioassay, air sampling, or contamination measurements,
- Physical form of the radionuclide from air sampling measurements, retention patterns, or process knowledge,
- Chemical form of the radionuclide from air sampling, contamination measurements, retention patterns, or process knowledge,
- Route of exposure from incident investigations or process knowledge,
- Previous exposure history, including environmental exposures,
- Sampling time after exposure from incident investigations,
- Bioassay information available including routine and/or special bioassays,
- Age and health status of exposed individual,
- Appropriate metabolic model from ICRP or NCRP recommendations and/or retention patterns, and
- Appropriate model to estimate intake, dose equivalent rate, or committed dose equivalent from ICRP or NCRP recommendations and/or retention patterns.

Most of the required information will be ascertained from the radiological incident investigation process described in Section 7.0 of this TBD. However, there may be instances when an exposure incident occurs undetected. Assumptions would then be based on facility process knowledge. In the absence of other data RPID will make conservative assumptions about the intake. These include:

- Inhalation exposure pathway,
- Conservative pulmonary clearance class, and
- One micron particle size.

There are two approaches to estimate when the exposure event occurred. The first conservatively assumes that the undetected event occurred directly after the last bioassay measurement. The time used to evaluate the intake would then correspond to the routine bioassay interval. The second approach, recommended by the ICRP, assumes the exposure event to occur at the mid-point of the bioassay monitoring interval. This approach provides an approximate acute intake solution for chronic exposures to the radionuclide compound (ICRP 88). The RPID Project considers undetected exposures to most likely have occurred from low-level chronic exposure and adopts the ICRP approach to assess undetected exposures. Please note that significant workplace controls exist to minimize chronic exposures, and RPID assumes that any internal exposures of significance will be acute.

The magnitude and dosimetry of all confirmed intakes in the RPID Project will be assessed. In addition, suspected intakes may be evaluated to ascertain the need for exposure assessment bioassays, exposure mitigation, and/or medical intervention.

6.2.1 Radionuclide Intake Assessment

Radionuclide intakes can be directly measured using *in-vivo* bioassay techniques or indirectly measured using *in-vitro* bioassay techniques. The intake can then be estimated from the bioassay data using appropriate metabolic models for the deposition and retention of the radionuclide compound. The following sub-sections describes the methodology for intake assessments in the RPID Project.

Tritium Intake Assessment

Regardless of the exposure pathway, the absorbed HTO is uniformly distributed throughout the whole-body in the form of body water. The rapid distribution of HTO throughout the body leads to equilibrium concentrations shortly after exposure. Within two hours of exposure, urine activity reflects the HTO concentration within the body. The retention of HTO, following an acute exposure, is characteristic of the normal turn-over rate of water and can be described using a single compartment model.

$$C(t)_{HTO} = C_0 e^{-\lambda_1 t}$$

where: $C(t)$ = HTO concentration in body water at time (t)

C_0 = Initial HTO concentration in body water

λ_1 = Removal rate of body water (d^{-1})

t = post acute exposure time interval (d)

. The removal rate of HTO in body waters can be calculated as follows:

$$\lambda_1 = \frac{0.693}{T_R} + \frac{0.693}{T_B}$$

where: T_R = Radiological half-life of tritium (4490d)

T_B = Biological clearance half-life of body water (assume 10d)

$\lambda_1 = 0.0693$ = Removal rate of body water (d^{-1})

The rapid uniform distribution of HTO throughout the body allows the RPID Project to assume that the HTO concentration in urine is the same as the concentration in the body at the time of voiding. Using this assumption along with an assumed body water volume of 42 liters, the equation given above is rearranged to solve for the initial intake of HTO.

$$I = 42 \bullet U(t) \bullet e^{0.0693 \bullet t}$$

where: I = Initial intake of HTO(pCi)

U(t) = Tritium concentration in urine at time t (pCi/L)

t = time post acute exposure (d)

HTO is assumed to be totally eliminated from the body in urine. Therefore, urine HTO concentrations are representative of body water HTO concentrations

Uranium and Plutonium Compound Intake Assessments

Intake assessments for uranium and plutonium compounds involve exposure pathway models (e.g., Lung and GI Models) and IRF and excretion functions containing multiple compartment functions. The complexity of these functions substantially increases the difficulty of intake assessment for these compounds. Several computer programs have been developed which assess intakes from bioassay measurements based on ICRP principles. The RPID Project has adopted the *INDOS* program to perform intake assessments. The following inputs are required to estimate the intake from uranium or plutonium compounds:

- Exposure Information; exposure pathway (e.g., inhalation, ingestion, and/or wound), exposure mode (e.g., chronic vs acute),
- Radionuclide Compound Information; isotope half-life, compound solubility class, particle size (inhalation exposures),
- Biokinetic Information; biological half-lives and compartment partitioning values, fractional excretion pathway values, and
- Bioassay Information; bioassay measurement type (e.g., urinalysis, WBC, etc.), measurement values, measurement time post exposure,

Section 3.0 of this TBD contains most of the information required by the *INDOS* program. Any required additional information should be obtained from the radiological incident investigation.

Intake Assessment for Other SNL COPC

The *INDOS* program is also capable of assessing the intakes of other SNL COPC. The program requirements are similar to the topics previously discussed. Radionuclide specific input parameters for secondary SNL COPC can be found in Appendix B of this TBD.

6.2.2 Intake Assessment from Nasal Smears

Please note that the deposition of radionuclides in nasal passages are not well understood and detailed biokinetic models are unavailable to reliably quantify intakes from nasal smears. The following method is used only for preliminary estimates of intake. The inhalation intake may be approximated using the following relationship:

$$I_{inh,i} = \frac{N_i}{e_{N-P} \bullet D_{N-P}}$$

where: N_i = Nasal smear activity of the i th radionuclide compound (Bq)

e_{N-P} = Nasal-pharyngeal collection efficiency

D_{N-P} = Nasal-pharyngeal deposition fraction

$I_{inh,i}$ = Inhalation exposure of the i th radionuclide compound (Bq)

The fraction of the total nasal-pharyngeal deposition collected on nasal smears has been observed to range between 8 and 15 percent (EGG 88). A value of 10 percent is considered to be reasonably conservative for intake estimates (EGG 88). The ICRP Lung Model provides nasal-pharyngeal deposition fractions for various particle AMAD. The ratio for 1 micron AMAD particles is 0.3. Inserting these values yields the following relationship for 1 micron AMAD particle exposures:

$$I_{inh,i} = 33.3 \bullet N_i$$

The total nasal smear activity for both nostrils should be used in these relationships. Intake estimates from nasal smears are more uncertain than other bioassay types. Intakes of small particles may not result in measurable nasal activity, while intakes of very large particles may lead to intake over-estimations. Other sources of uncertainty are whether the individual is a "mouth breather" or has a cold or sinus condition. Finally, nasal smears can be easily cross-contaminated. Vastly different right and left nostril measurements strongly suggest external contamination rather than inhalation (EGG 88).

6.2.3 Multiple Exposure Intake Assessment

The presence of pre-existing radionuclide compound depositions, including environmental exposures, can yield erroneous results when not compensated. Therefore, bioassay measurements will be adjusted for pre-existing exposures. The 10 CFR 835 dose limits pertain to occupational exposures only. The dosimetry bioassay measurement for radionuclides involved in previously known exposures is estimated as follows:

$$DBM = CBM - \sum_N EBM$$

where: CBM = Confirmed bioassay measurement (pCi/L)

EBM = Estimated bioassay measurement for N previous exposures (pCi/L)

DBM = Dosimetry bioassay measurement (pCi/L)

The estimated bioassay measurements are derived using the INDOS program using known previous intake histories (e.g., initial body burden, time of exposure, etc.) as inputs.

Although modeled as such, the removal of a radionuclides from the body is not constant and is subject to fluctuations over time. Potential additional exposures should be carefully reviewed to determine if additional evidence (e.g., air sampling, contamination monitoring, etc.) support an additional exposure hypothesis. The variance of past measurements from predicted values should also be considered. The ultimate decision regarding whether bioassay findings represent new or previous documented exposures will be made by the Internal Dosimetry Project Leader.

6.2.4 Estimating Intake from Multiple Bioassay Measurements

Several exposure assessment bioassays are expected to be obtained for reportable intakes in the RPID Project. Intake estimates from individual bioassay measurements are typically highly variable. This variability does not necessarily invalidate the bioassay measurements since metabolic fluctuations are expected. The *INDOS* program includes a fitting function which calculates the "best fit" intake from multiple bioassay measurements. The iterative weighting option is recommended since this technique is not greatly influenced by abnormally high or low bioassay results (Skrable 86). The *INDOS* iterative weighting fit function will be used for multiple bioassay assessments to evaluate intakes in the RPID Project. The intake estimate will be constantly refined as addition bioassay measurements are made.

6.2.5 Dosimetry Assessment

Dosimetry assessments are performed in the RPID Project by multiplying the initial intake(I) estimated from bioassays by the dose conversion factor (DCF) for the detected radionuclide. The CEDE from the detected exposure can then be assessed as follows:

$$H_{50,wb} = \sum_i I_i \bullet DCF_{wb,i}$$

where: I_i = Estimated intake of the i^{th} radionuclide(uCi)

$DCF_{wb,i}$ = Stochastic dose conversion factor for the i^{th} radionuclide(rem/uCi)

$H_{50,wb}$ = Committed effective dose equivalent for the evaluated exposure event (rem)

The Environmental Protection Agency (EPA) Federal Guidance Report 11 *Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion* contains pathway and radionuclide compound specific based on the ICRP 30 methodologies (EPA 88). This and similar references can be used as DCF sources. The CEDE will be assessed for all detected exposures in the RPID Project.

6.2.6 Critical Organ Dose Assessment

Intakes for certain radionuclide compounds (e.g., radioiodines, bone seekers, etc.) may be limited by the non-stochastic provisions in the 10 CFR 835. For the purposes of this TBD, the critical organ is defined as the organ or tissue for which the non-stochastic limit is based (e.g., thyroid for radioiodines, bone surface for Class Y plutonium, etc). The committed effective dose from the detected exposure can then be assessed as follows:

$$H_{50,T} = I_i \bullet DCF_{T,i}$$

where: I_i = Estimated intake of the i^{th} radionuclide(uCi)

$DCF_{T,i}$ = Non-stochastic dose conversion factor for the i^{th} radionuclide(Rem/uCi)

$H_{50,T}$ = Committed dose equivalent for the evaluated exposure event (rem)

The EPA Report 11 also contains non-stochastic DCF for numerous radionuclide compounds from the inhalation and ingestion exposure pathways (EPA 88). The committed dose equivalent will be assessed for all critical organs whenever dose is assessed in the RPID Project.

6.2.7 Assessing Total Effective Dose Equivalent

The stochastic and non-stochastic regulatory dose limits are based on the TEDE. The TEDE is the combined annual dose from internal and external radiation sources. The TEDE for N intakes of radionuclide compounds during the calendar year is calculated as follows:

$$TEDE = \sum_N H_{50,wb,i} + DDE_{ext}$$

where: $H_{50,wb,i}$ = The CEDE from the i^{th} radionuclide intake (rem)

DDE_{ext} = Deep dose equivalent from external radiation sources (rem)

TEDE = Total effective dose equivalent (rem)

Monitoring TEDE requires an interphase between the SNL Internal Dosimetry and External Dosimetry Projects. The Sandos computer database contains the results of both external and internal monitoring. These results are summed so that dose controls are based on TEDE exposure.

6.2.8 Adjusting Dose Estimates for Different AMAD Particles

The ALI and DCF are calculated for inhalation exposures to 1 micron AMAD particles. Different particle size distributions may significantly alter the inhalation exposure dosimetry. The following relationship can be used to adjust the 1 micron AMAD DCF (ICRP 79):

$$\frac{H_{50,T}(AMAD)}{H_{50,T}(1\mu m)} = f_{N-P} \bullet \frac{D_{N-P}(AMAD)}{D_{N-P}(1\mu m)} + f_{T-B} \bullet \frac{D_{T-B}(AMAD)}{D_{T-B}(1\mu m)} + f_P \bullet \frac{D_P(AMAD)}{D_P(1\mu m)}$$

where: f = Fraction of committed dose equivalent in the reference tissue from deposition in the nasal-pharyngeal (N-P), tracheal-bronchial (T-B), and

pulmonary (P) regions.

D = Deposition probabilities for the measured (AMAD) and standard (1 μm) particle size AMAD values for the N-P, T-B, and P regions.

$H_{50,T}$ = The dose conversion factor for the reference tissue from exposure to the measured (AMAD) and standard (1 μm) AMAD values.

Figure 6-1 demonstrates the ICRP Lung Model regional deposition probabilities for various particle size AMADs. Values for the fractional committed dose equivalents in the ICRP pulmonary compartments for various particle sizes can be found in (**reference**). The CEDE can then be determined by summing the weighted individual effective doses for all of the organs or tissues specified in Table 6-2.

The ICRP recommends that a value of 1 micron AMAD be assumed for unknown particle size distributions (ICRP 79). This assumption is considered sufficiently conservative for most dosimetry applications. Therefore, the RPID Projects adopts the 1 micron assumption. Particle size adjustments will be made at the discretion of the Internal Dosimetry Project Leader.

6.2.9 Calculating Total Effective Dose Equivalent for Pregnant Workers

Exposures to internal radionuclide materials which may be transferred to an embryo/fetus must be estimated to ensure radiation protection standards for declared pregnant workers are maintained. These calculations are more complicated since placental transfer biokinetics must be included and embryonic/fetal development is in constant state of change which subsequently alters physical geometries, mass, and embryo/fetal biokinetics. *Contribution of Maternal Radionuclide Burdens to Prenatal Radiation Doses* contains dosimetry guidelines which can be used to provide estimates of embryo/fetal doses for regulatory purposes (NRC 92). Embryo/fetus exposure can result from the following sources:

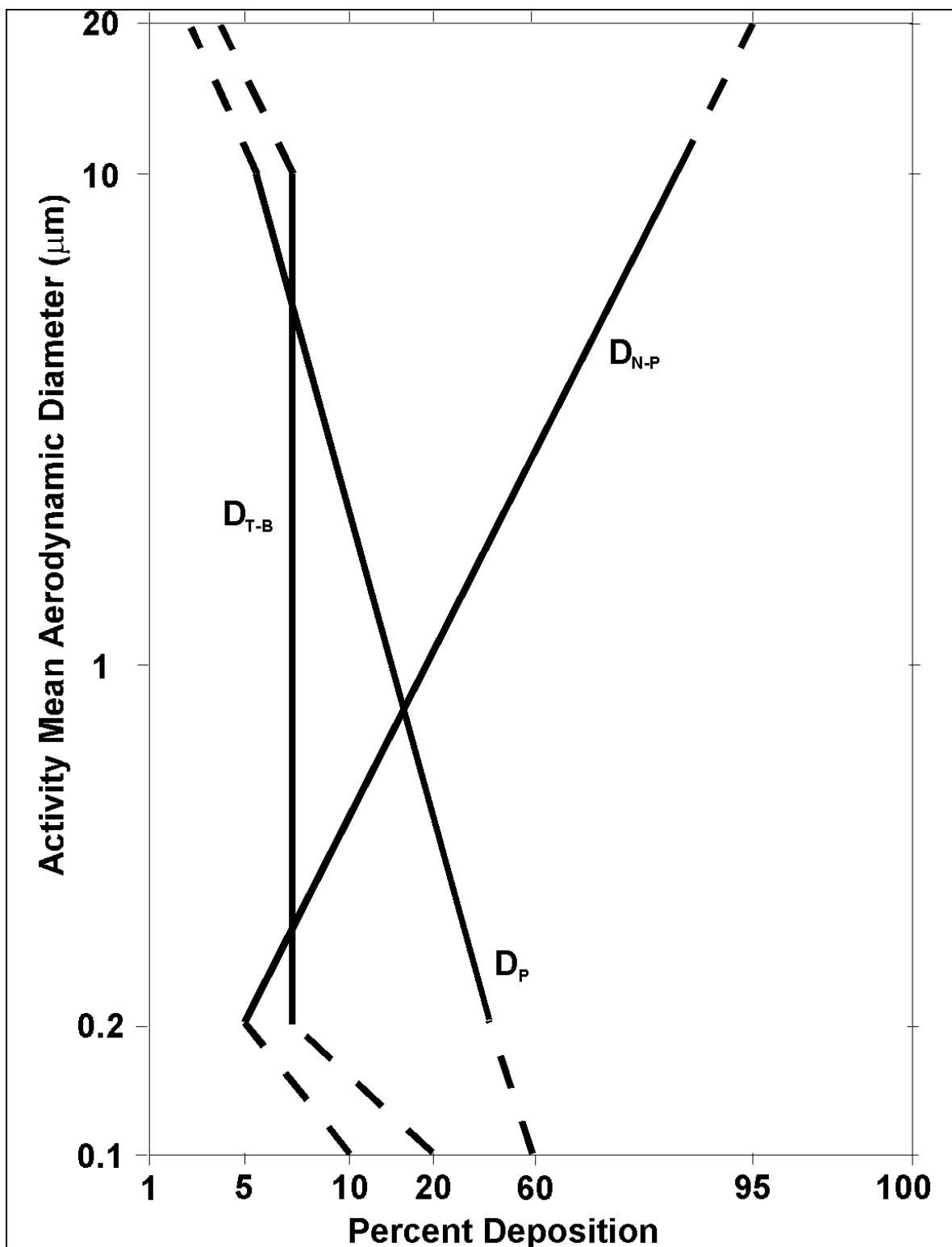


Figure 6-1 ICRP Lung Model Deposition Fractions

- External sources from outside of the maternal body,
- External sources from radionuclides deposited within the maternal body, and
- Internal sources originating from maternal systemic circulation which transferred radionuclides to the embryo/fetus via the placenta.

Two techniques are described in *Contribution of Maternal Radionuclide Burdens to Prenatal Radiation Doses* which estimate exposure from maternal internal radionuclide compound exposures (NRC 92). The first technique uses biokinetic transfer models based on limited human and animal data. Precursory models have been developed for the following radionuclide compounds:

- Inorganic Tritium and Carbon-14,
- Organic Tritium and Carbon-14,
- Cobalt compounds,
- Strontium compounds,
- Ruthenium compounds,
- Iodine compounds,
- Cesium compounds,
- Uranium compounds,
- Plutonium compounds, and
- Americium compounds.

Fetal exposure is then evaluated using these models combined with MIRD techniques. Exposure is evaluated on a monthly basis due to the temporal changes in embryonic/fetal development and is a function of when the intake occurred. The dosimetry assessment technique is outlined as follows:

- Determine the date of intake and approximate stage (days) of gestation at intake,
- Compute maternal radionuclide burden (i.e., from *in-vitro* and *in-vivo* analytical results),
- Estimate the monthly radiation dose to the embryo/fetus Appendix D of *Contribution of Maternal Radionuclide Burdens to Prenatal Radiation Doses* (NUREG 92), and
- Compute the integral dose for the entire pregnancy by combining monthly estimates.

The available data for this technique includes all of the primary SNL COPC and many of the potential secondary COPC. This dosimetry technique is the current state-of-art method and is considered sufficiently conservative for use in the SNL Internal Dosimetry Project for exposures to the published radionuclide compounds.

The second technique involves approximating embryonic/fetal dosimetry from uterine doses. Uterine doses are calculated using MIRD techniques and do not consider the temporal changes of the embryonic/fetal environment. Estimates using this approach are 50 year committed doses compared to the monthly exposure estimates using the first approach. Therefore, this technique

can significantly over-estimate embryo/fetal dose, especially for radionuclide compounds with long retention times. Table 6-3 compares dosimetry estimates between the two approaches (i.e., term dose estimate from the biokinetic models versus the committed dose estimate using the surrogate uterine dose technique) (NRC 92).

Radionuclide	Biokinetic/Uterine Dose Approach Ratio	Radionuclide	Biokinetic/Uterine Dose Approach Ratio
Tritium (3H)	0.14 to 0.92	Iodine-131	0.40 to 13.04
Carbon-14	0.09 to 0.62	Cesium-134	0.39 to 1.34
Cobalt-57	0.14 to 0.29	Cesium-137	0.30 to 1.11
Cobalt-58	0.17 to 0.26	Uranium-234	0.15 to 3.06
Cobalt-60	0.10 to 0.27	Uranium-235	0.14 to 2.87
Strontium-89	1.77 to 6.20	Uranium-238	0.15 to 2.43
Strontium-90	0.90 to 2.80	Plutonium-238	1.68 to 34.11
Ruthenium-106	0.02 to 0.06	Plutonium-239	1.72 to 28.19
Iodine-125	0.76 to 12.69	Americium-241	0.10 to 2.00

Table 6-3 Comparison of Embryo/Fetal Dosimetry Techniques

The ratio range reflects the variable monthly dose estimates provided by the biokinetic model approach. The most soluble compound is assumed in the calculations in Table 6-3. The advantage of the uterine surrogate dose approach is estimates of all radionuclides can be made without the extensive research required for the biokinetic approach. Therefore, this approach may be used for exposures which can not be evaluated using the biokinetic approach. Appendix F in *Contribution of Maternal Radionuclide Burdens to Prenatal Radiation Doses* contains embryo/fetal dose estimates for several radionuclides based on the surrogate uterine dose approach (NUREG 92). The uncertainties associated with surrogate uterine dose approach can then be reduced using the values in table 6-1 for radionuclides with similar emission and biokinetic properties as follows:

$$D_f = D_U \bullet R_{B/U}$$

where: D_U = Surrogate uterine dose estimate (rad)

$R_{B/U}$ = Ratio between biokinetic and surrogate uterine dose models

D_F = Embryonic/fetal dose (rad)

The ratio for americium compounds is recommended for all transuranic radionuclides not specifically evaluated by the biokinetic approach (NRC 92). The RPID Project will use the largest ratio value when necessary to provide conservative fetal/embryonic dose estimates.

The embryo/fetal doses estimated by both techniques are expressed in terms of absorbed dose (i.e., rad). The dose equivalent can be obtained by multiplying the absorbed dose by the appropriate factor in Table 6-1.

6.2.10 Dose Assessment Using Multiple Bioassay and Measurement Techniques

Exposures from some radionuclide compounds may be quantified using multiple bioassay and measurement techniques (e.g., urinalysis and WBC for ^{137}Cs exposures). The subsequent estimated intake may be different because of the unique uncertainties inherent in each bioassay and/or measurement technique. In this case, the dosimetry assessment should be based on the bioassay and measurement technique which is considered the least uncertain. Consideration should be given the following when selecting the most accurate bioassay technique:

- Analytical sensitivities; measurements near the LLD are more uncertain,
- Measurement uncertainties; gross analytical methods are less accurate than isotopic analyses,
- Biokinetic model uncertainties; *in-vitro* measurements require more assumptions than *in-vivo* measurements, and
- Measurement variability; bioassay techniques which show high variability between measurements may be more uncertain.

The determination of which bioassay method is most accurate is subject to the professional judgement of the Internal Dosimetry Project Leader. Regardless of the assessed accuracy, all bioassay techniques which yield positive results will be continued until the the RPID project leader determines that sufficient data is available to accurately quantify the intake.

6.3 Personalized Dose Assessment

Dose assessments based on the ICRP methodology contain numerous assumptions based on the "Reference Man" principle. However, considerable variability exist between individuals which can significantly alter internal exposure dosimetry. This Section provides guidance on identifying the need and developing personalized dose assessments. The need for developing individual IRF should be re-evaluated whenever additional bioassay information becomes available.

6.3.1 Determining the Need for Personalized Dose Assessment

The ICRP recommends that individual IRF be developed when the individual dose estimates differ by 50 percent or greater and when an estimated potential exposures exceeds 1 rem (ICRP 88). Dose estimates between various bioassay and measurement techniques are expected to differ (e.g., WBC and urinalysis, gross and isotopic analyses). However, individual IRF will be developed in the RPID Project when dosimetry estimates using similar bioassay and measurement techniques exceeds the ICRP criteria. Individual IRF will be required when the following conditions are not met:

$$\frac{\sigma}{\mu} < 0.5;$$
$$\mu + \left(t_{95\%} \cdot \frac{\sigma}{\sqrt{N}} \right) < 1rem$$

where: σ = Standard deviation of N dosimetry assessments (rem)

μ = Mean dosimetry assessment for N measurements (rem)

$t_{95\%}$ = Normal distribution t statistic value at the 95% confidence interval

N = Number of bioassay measurements

The 95 % normal distribution t statistic value is dependent on the number of bioassay measurements (N). Table 6-4 contains values for $t_{95\%}$ for several values of N:

Number of Bioassays (N)	$t_{95\%}$	Number of Bioassays (N)	$t_{95\%}$
2	6.314	7	1.943
3	2.920	8	1.895
4	2.353	9	1.860
5	2.132	10	1.833
6	2.015	11	1.812

Table 6-4 Normal Distribution t Statistic Values at the 95% Confidence Level

It should be noted that each dosimetry assessment involves multiple bioassay measurements which were "fitted" by the *INDOS* program. The need for personalized IRF will be evaluated whenever the "fitted" intake estimate is updated.

The bioassay data representativeness should be evaluated whenever the ICRP criteria are exceeded. The biokinetic clearance of radioactive materials shortly after an exposure event can be highly variable and is not well characterized by ICRP models. Consideration should be given to not using early bioassay data when assessing the ICRP criteria. Bioassay measurement trends

should also be studied to identify the presence of unusually high or low results. The representativeness of identified spurious measurements should be evaluated. Bioassays which are not considered to be representative will be removed from the dose assessment. The omission of any bioassay data is subject to the Dosimetry Project Leader's approval.

6.3.2 Re-Assessment of Dosimetry Assumptions

Dose assessments require a number of parameters including time of exposure, particle size AMAD, and chemical form. These parameters are often based on process knowledge which may not accurately reflect the exposure conditions. Therefore, dose assessments should be re-evaluated using alternate values unless samples representative of exposure conditions exist. The combination of assumptions which best fit the bioassay data is then adopted. Personalized IRF will be developed if the re-evaluated dose assessments fail to meet the ICRP criteria.

6.3.3 Developing Personalized Intake Retention Functions

Intake retention functions are typically developed in the following form:

$$IRF(t) = \sum_{i=1}^N a_i e^{-\lambda_i \cdot t}$$

where: a_i = Fraction of body burden metabolized in the i th compartment

λ_i = Biokinetic removal constant for the i th compartment (d^{-1})

t = time post exposure (d)

$IRF(t)$ = Fraction of initial intake present at time t

The first compartment (i.e., $i = 1$) represents the initial rapid clearance component which is considered to have a 0.25 day half-life by the ICRP. This representation is highly uncertain since the rapid clearance component is often poorly understood. However, this component does not significantly influence the IRF for bioassays collected greater than two days after the acute intake event. Therefore, the rapid clearance component will be ignored for the purpose of developing personalized IRF. The bioassay data influenced by the rapid clearance component will not be used to develop personalized IRF.

Intake retention functions consisting of two terms (i.e., one compartment in addition to the rapid clearance component) can be developed using a least squares technique fitting the natural logarithm transform of bioassay data versus time. The least square method minimizes the error of a linear fit to the data. Computer algorithms for least square fitting techniques can be found in several references including *Numerical Analysis* (Burden et al., 81). The RPID Project possesses the computer software *Mathematica* which is capable of least square approximations (Wolfram 91). This program will be used to develop personal IRF for the RPID Project.

Intake retention functions containing more than three terms require complicated non-linear fitting techniques that provide solutions which often can not be directly used by *INDOS*

computer program. A potential solution may result by simplifying these IRF. Several IRF contain a very long biologic removal component which can be effectively considered constant. Personalized IRF which meet the 10 CFR 835 requirements may be developed using least square techniques using this assumption. The *Mathematica* program will be used to provide non-linear fitting techniques when the least square fit does not meet regulatory guidelines.

The ultimate intake estimated from personalized IRF will be obtained using the following relationship:

$$I_p = \frac{I_{p-IRF}}{(1 - a_1)}$$

where: I_{p-IRF} = Intake estimated using the personalized IRF (Bq)

a_1 = Fraction of the body burden metabolized in the rapid clearance compartment

I_p = Final intake estimate using personalized IRF (Bq)

This relationship compensates for the rapid clearance component which was not included in the personalized IRF.

6.4 Dosimetry Based on Air Sampling

The 10 CFR 835 requires all dose assessments to be based on bioassay data only. Exposures which yield positive air and bioassay samples will be evaluated using only the bioassay measurements. However, the bioassay sensitivity for certain radionuclide compounds is not sufficient to detect reportable exposures defined in the 10 CFR 835 (i.e., a technology shortfall exists). Dose may be assessed from air sampling results in this case (10 CFR 835). Therefore, all detected air concentrations of radionuclide compounds with technology shortfalls in the SNL Air Sampling Project will be reported to the Internal Dosimetry Project Leader. The following radionuclide compounds are considered to possess technology shortfalls in the RPID Project:

- Class W and Y plutonium isotopes,
- Class Y uranium isotopes,
- Class Y ^{232}Th and ^{228}Th ,
-

Air sampling measurements are expressed in terms of DAC-hours which are calculated as follows:

$$DAC - h = \frac{C}{DAC} \bullet T_E$$

where: C = Average measured air concentration (uCi/m³)
 DAC = Derived air concentration (uCi/m³)
 T_E = Time of exposure (h)
 DAC-h = pseudo inhalation exposure quantity (h)

Representative air samples, expressed in terms of DAC-h, can be converted into terms of intake or dose using the following relationships:

$$I = \frac{DAC-h}{2000} \bullet ALI;$$

$$CEDE = \frac{DAC-h}{2000} \bullet 5$$

where: I = Radionuclide intake (uCi)
 DAC-h = pseudo inhalation exposure quantity (h)
 ALI = Annual limit of intake of the detected radionuclide compound (uCi)
 CEDE = Inhalation event committed effective dose equivalent (rem)

Inhalation exposures to mixtures of radionuclide compounds can be assessed by summing the individual dose components.

Dose assessments from air samples are highly uncertain because of the difficulty of collecting air samples which accurately reflect what the worker is breathing. Air sampling devices are characterized into three general classes; general area, breathing zone, and personal air samplers. Personal air samplers (e.g., lapel samplers) are considered to be the most representative since these devices sample air near the worker's nose and mouth (NRC 1991). Therefore, dose assessments in the RPID Project will be limited to personal air sampler measurements.

Care should be taken when dose is assessed from air samples. Airborne release events should be collaborated with positive measurements from other air samplers located in the facility. Contamination of the air sample from external sources should be ruled out. All measurements should be compensated for the presence of naturally occurring radon and thoron progeny. In addition, air sample results which indicate exposures which should be detected via bioassay should be considered invalid. All recorded doses calculated from air sampling results in the RPID Project will be noted as such.

6.5 Dosimetry Quality Control

Dosimetry quality control is mandated by the 10 CFR 835 and is necessary to ensure accurate dose assessments.

7.0 Investigation Levels and Radiological Incident Response

Radiological controls at SNL/NM are designed to prevent or minimize radiological incidents. Therefore, radiological incidents are expected to be rare events. Prompt evaluations of incident bioassays are prudent whenever radiological incidents occur. This Section establishes the SNL/NM Internal Dosimetry Program response to radiological incidents. Initiating events are identified and the response to various radiological incidents are specified.

7.1 Investigation Initiating Events

The appropriate response to a radiological incident is dependent on the severity of the potential exposure. Derived investigation levels (DIL) are used in the SNL/NM Internal Dosimetry Program to identify both reportable exposures (i.e., exceeding 100 mrem CEDE) and exposures potentially requiring medical intervention (i.e, 5 rem CEDE). DIL are radionuclide specific and are developed for the inhalation exposures only since this is the most probable exposure pathway at SNL/NM.

7.1.1 Reportable Exposure Derived Investigation Levels

Reportable events are defined in 10 CFR 835 as radiological incidents resulting in exposures exceeding 100 mrem CEDE. The DIL for reportable exposures is defined as the bioassay measurement corresponding to a 100 mrem CEDE. The urinary excretion function for a radionuclide compound is described as follows:

$$E(t)_U = \frac{U(t)}{BB(0)} = f_U \cdot \left(\frac{d}{dt} \sum_{i=1}^N a_i e^{-\lambda_i t} \right)$$

where: $U(t)$ = Total 24 hour urinalysis measurement at time t (uCi)

$BB(0)$ = Radionuclide compound intake (uCi)

f_U = Fraction of body burden excreted in urine

a_i = Fraction of body burden metabolized in the i th compartment

λ_i = Effective biological removal constant = $0.693/t_e$ (d^{-1})

t = time post exposure (d)

A reportable incident is equivalent to 2 percent of the radionuclide specific ALI. Differentiating the IRF and setting the urinalysis measurement equal to the DIL yields:

$$DIL_U = 0.02 \cdot ALI \cdot f_U \cdot \left(\sum_{i=1}^N \lambda_i \cdot e^{-\lambda_i t} \right)$$

where: ALI = Annual limit of intake (uCi)

DIL_U = Total 24 hour urinalysis derived investigation level (uCi)

This function describes the retention of radionuclide compounds which have been absorbed into the systemic circulation. The excretion of rapidly absorbed compounds (e.g., tritium) can be approximated with this expression. However, additional consideration must be given to exposure pathway biokinetics (i.e., ICRP Lung and GI Tract models) for less soluble radionuclide compounds. DIL can be determined for *in-vivo* bioassays as well as *in-vitro* bioassays when appropriate.

The initial 24 hour urinalysis measurement after a radiological incident is subject to the IRF rapid clearance component. The biokinetics of this component is poorly understood and not accurately modeled using ICRP methodologies. Therefore, the DIL will be based on the second day post exposure measurement (i.e., $t = 2$) since rapid clearance effects are reduced.

Intakes from chronic low-level concentrations may be difficult to detect for some airborne radionuclide compounds. The timing of the exposure event may be difficult, or impossible, to ascertain. Assuming a single acute exposure occurring at the mid-point of the bioassay monitoring interval approximates low-level chronic exposures in these cases (ICRP 88). The DIL for undetected radiological incidents will be developed under this assumption in the SNL/NM Internal Dosimetry Program.

7.1.2 DIL for Tritium

The inhalation ALI for HTO is 81.1 mCi (3×10^9 Bq). Using this value in conjunction with the HTO IRF yields a DIL of 98.1 uCi per 24-hour urine sample at 2 days post exposure. The "Reference Man" daily urinary output is 1.4 L (ICRP 75). The corresponding DIL urinary concentration at 2 days post exposure is 70.1 nCi/ml. The routine bioassay monitoring frequency for HTO is 60 days. The corresponding DIL for undetected radiological incidents is 6.2 nCi/ml.

7.1.3 DIL for Uranium and Plutonium Compounds

Uranium and plutonium compounds encountered at SNL/NM are expected to be in relatively insoluble forms. Absorption into systemic circulation is dependent on biokinetics of these compounds within the lung. Therefore, the DIL for these compounds will be developed using the *INDOS* computer program which is based on the ICRP Lung Model. Table 7-1 contains the detected and undetected radiological incident DIL for uranium and plutonium compounds.

Radionuclide	ALI (Bq)	LLD (pCi/ml)	Detected Incident DIL (pCi/ml)	Undetected Incident DIL (pCi/ml)
<u>Uranium-234</u> Class D Class W Class Y	50,000 30,000 1,000	0.1	0.8 (pos) ¹ (pos)	0.17 (pos) (pos)
<u>Plutonium-238</u> Class W Class Y	200 500	0.06	(pos) (pos)	(pos) (pos)
<u>Plutonium-239</u> Class W Class Y	200 600	0.06	(pos) (pos)	(pos) (pos)
<u>Americium-241</u> Class W	200	0.06	(pos) (pos)	(pos) (pos)

¹ Any positive measurement would exceed the DIL

Table 7-1 Derived Investigation Levels for Uranium and Plutonium Compounds

Table 7-1 demonstrates that almost all confirmed positive measurements for uranium and plutonium compounds would exceed the DIL.

7.1.4 DIL for other SNL/NM COPC

DIL for detected and undetected radiological incidents for other SNL/NM COPC will be developed using the same methodology described in this Section. Appendix B contains DIL summary tables for undetected radiological incidents for these compounds.

7.2 Medical Intervention Levels

Medical intervention levels (MIL) are derived to quickly identify bioassay measurements which may indicate the need for medical intervention to mitigate exposure. The 10 CFR 835 annual dose limit for radiological workers is 5 rem CEDE. Restrictions may be placed on radiological workers who exceed the annual limit. Therefore, medical intervention should be considered whenever an estimated exposure from a single acute exposure event exceeds 5 rem CEDE. The MIL is directly proportional to the DIL since dose varies linearly with intake. The MIL can be estimated using the following relationship:

$$MIL = 50 \bullet DIL$$

where: DIL = Derived investigation level (pCi/ml)
MIL = Medical intervention level (pCi/ml)

The MIL for detected and undetected exposures to tritiated compounds are 3.5 uCi/ml and 0.3 uCi/ml respectively. Table 7-2 contains the MIL for uranium and plutonium compounds.

<u>Radionuclide</u>	ALI (Bq)	LLD (pCi/ml)	Detected Incident MIL (pCi/ml)	Undetected Incident MIL (pCi/ml)
<u>Uranium-234</u>				
Class D	50,000		39.5	8.7
Class W	30,000	0.1	3.2	1.2
Class Y	1,000		(pos) ¹	(pos)
<u>Plutonium-238</u>				
Class W	200	0.06	(pos)	(pos)
Class Y	500		(pos)	(pos)
<u>Plutonium-239</u>				
Class W	200	0.06	(pos)	(pos)
Class Y	600		(pos)	(pos)
<u>Americium-241</u>				
Class W	200	0.06	(pos)	(pos)

¹ Any positive measurement would exceed the MIL

Table 7-2 Medical intervention levels for Uranium and Plutonium Compounds

Appendix B contains the undetected radiological incident MIL for other SNL/NM COPC.

7.3 Radiological Incident Response

An internal dosimetry investigation will be initiated for all reportable events (i.e., internal dose exceeding 100 mrem CEDE). The internal dosimetry investigation involves the following:

- Re-analyze bioassay sample,
- Evaluate bioassay quality,
- Contacting effected parties,
- Initiate incident bioassay monitoring
- Meeting with effected parties,
- and, Re-evaluating dose assessment.

The laboratory yielding the positive bioassay result should be promptly contacted. All laboratory calculations should be verified. Split samples should be analyzed to validate the

positive finding when available. However, these measurements may not be necessary when the positive findings are significantly above the analytical quantification level (L_Q) as defined by Currie (Currie 68). The quality of the analytical data should be re-scrutinized. Further actions may be delayed by the Internal Dosimetry Project Manager if data quality is suspect.

Next, the radiological worker will be notified of the positive bioassay. Other responsible parties will also be notified. Table 7-3 contains the responsible parties and various required time frames as a function of the initial dose assessment:

Initial Dose Assessment (rem)	Parties to be Contacted	Time frame
$0.1 \leq D < 0.5$	Exposed Individual Immediate Supervisor Department 7714 Manager	5 days
$0.5 \leq D < 2$	Exposed Individual Immediate Supervisor Department 7714 Manager SNL/NM Management	5 days
$2 \leq D < 5$	Exposed Individual Immediate Supervisor Department 7714 Manager SNL/NM Management DOE Management	2 days
$D > 5$	Exposed Individual Immediate Supervisor Department 7714 Manager SNL/NM Management DOE Management Occupational Health Center	As soon as possible

Table 7-3 Dosimetry Investigation Required Contacts and Time Frames

Once contacted, the effected parties will be advised on the assessed dose, the need for incident bioassays, and the need for medical intervention. A internal dosimetry investigation meeting will be arranged between all effected parties. Topics to be discussed during the internal dosimetry investigation meeting include the following:

- Events leading to the internal exposure,
- Potential physical and chemical forms of all radionuclide compounds involved,
- Initial dosimetry assessment,
- Significance of assessed dose,
- Need for medical intervention,
- Required incident bioassay program,
- Potential work limitations,
- Evaluation of radiological controls at the facility,
- Evaluation of individual work habits,

- and, Collection of additional information (e.g., swipe surveys, air monitoring results, etc.).

The conclusions and actions resulting from the internal dosimetry investigation meeting should be acceptable to all effected parties. The original dose assessment may be re-evaluated based on the conclusions of this meeting. Table 7-4 specifies the timing for the internal dosimetry investigation meeting.

Initial Dose Assessment, D (rem)	Required Internal Dosimetry Investigation Meeting Time Frame
$0.1 \leq D < 0.5$	10 days
$0.5 \leq D < 2$	3 days
$2 \leq D < 5$	1 day
$D > 5$	4 hours

Table 7-4 Required Internal Dosimetry Investigation Meeting Time Frames

A report will be generated which summarizes the findings and conclusions of the internal dosimetry investigation meeting. This report will be reviewed by the SNL/NM Internal Dosimetry Program Manager for completeness and accuracy. All effected parties will be provided copies of this report. Any changes to the report findings will be documented and revised copies will be provided to all effected parties.

7.4 Medical Intervention

Significant internal exposures are rare events at SNL/NM and are expected to result from accident events only. Prompt and appropriate action can minimize the internal contamination and diminish potential harmful effects resulting from large accidental intakes. Effective dose mitigation depends on co-operation between the health physics staff (Departments 7714 and 7715), Occupational Medicine Center (Department 3300) physicians and staff, line managers, and the exposed individual. Appropriate responses to radiological emergencies are prioritized as follows:

- Immediate medical care,
- External decontamination,
- and, Internal dose mitigation.

Internal exposures to radionuclide compounds are rarely immediately life threatening. Immediate emphasis is always placed on the treatment of any serious physical injuries associated with the accident event. Decontamination of the effected individual minimizes potential internal exposures and prevents the spread of contamination to un-controlled areas. Decontamination is

emphasized when minor or no injuries are involved. The use of medical intervention techniques is determined by a risk/benefit analysis which weighs the potential dose reduction versus the risks associated with the medical intervention technique. Such analyses require a thorough knowledge of the healing arts which is not within the scope of the SNL/NM Internal Dosimetry Program. Therefore, these evaluations will be made by qualified Department 3300 medical practitioners only.

The SNL/NM Radiation Emergency Procedure Manual describes the responsibilities of involved individuals/organizations and defines the approved medical responses to radiological emergencies (SNL 91). The selected medical intervention techniques are based on the recommendations within the NCRP Report *Management of Persons Accidentally Contaminated with Radionuclides* (NCRP 80). Internal Dosimetry Program personnel should consult these documents when assisting Department 3300 physicians.

There is presently no consensus when medical intervention techniques should be required among regulatory agencies, the NCRP, and the ICRP. The NCRP report *Basic Radiation Protection Criteria* considers planned doses up to 25 rem TEDE to be reasonably acceptable for emergency conditions (NCRP 71). It follows that accidental exposures up to the same level should not cause major concern (NCRP 80). For the purposes of this TBD, potential exposures exceeding 25 rem TEDE are considered serious. This corresponds to a maximum organ committed effective dose of 250 rem for intake limits based on non-stochastic effects. Medical intervention techniques are strongly indicated in the event of serious exposures. Medical intervention should be considered for exposures between the MIL (i.e., 5 rem CEDE or 50 rem committed effective dose) and 25 rem TEDE.

7.4.1 Internal Dosimetry Program Role in Medical Interventions

The primary role of the SNL/NM Internal Dosimetry Program in radiological emergencies is providing dosimetry support to Department 3300 physicians. The historical details of the radiological incident should be promptly collected and reviewed to ascertain the probable chemical and radiological composition of the internal exposure. Bioassay measurements should be made to determine the intake magnitude. However, initial intake assessment from bioassay measurements may be delayed, highly uncertain, or impossible due to the following:

- Early clearance biokinetics may lead to over estimation of the intake,
- Difficulty in collecting representative 24 hour *in-vitro* bioassay sample soon after a radiological incident,
- Presence of surface contamination may lead to over estimations of intake via *in-vivo* bioassay techniques,
- Significant inhalation exposures of insoluble compounds may not be detected in initial *in-vitro* bioassays,
- and, Laboratories may be incapable of providing prompt quantitative analytical results.

The effectiveness of medical intervention techniques is substantially improved when initiated promptly after the intake event. Therefore, medical intervention decisions may be based on limited superficial evidence such as air sampling, contamination surveys, and inventory assessments. The following sections discuss techniques which may be used to identify serious intakes (i.e., greater than 25 rem CEDE) in lieu of accurate bioassay data.

7.4.2 Identifying Potentially Serious Inhalation Exposures

Inhalation is considered to be the most probable exposure pathway at SNL/NM. Initial estimates of the potential magnitude of an inhalation exposure can be made by examining air sampling and nasal smear results, and/or estimating the hazard index of the release event.

Air Sampling Measurements

Properly placed air samplers may provide representative samples of the air breathed by the exposed individual. The relative inhalation hazard may be approximated by the following relationship:

$$IDI = 1.0 \times 10^{-4} \cdot \sum_{i=1}^N \frac{\overline{C_i} \cdot t_e}{DAC_i}$$

where: C_i = Average air concentration of the i th radionuclide compound during the time of exposure (uCi/ml)

t_e = Interval of time the individual was exposed to airborne contaminant (h)

DAC = Derived air concentration of the i th radionuclide compound (uCi/ml)

IDI = Inhalation dose index for exposure to N radionuclide compounds

Medical interventions should be seriously considered when the inhalation dose index exceeds unity. The average air concentration (C) can be determined from the measured activity divided by the total volume of air being sampled during the radiological incident (i.e., all activity is assumed to occur from the acute accident event). The highest uncertainty associated with using air samples is sample representativeness. The lowest uncertainties are achieved from air samples collected from the breathing zone of the monitored individual. Exposures may be over-estimated when the air sample is collected between the exposed individual and the release point, while under-estimations may occur when sampling air which has already passed the exposed individual. Rapid air sample analysis requires the ability to discriminate natural background radionuclides (e.g., radon progeny) from the COPC.

Nasal Smears

Positive nasal smears highly suggest that an inhalation exposure has occurred. A general rule of thumb for potential plutonium exposures is that measurements exceeding 500 dpm indicate potentially serious exposures, while measurements less than 50 dpm suggest no more than a possible low order exposure (NCRP 80). This rule of thumb will be adopted for all exposures to alpha emitting isotopes in the SNL/NM Internal Dosimetry Program. An inhalation dose index can be estimated when isotopic information is available using the following relationship:

$$IDI = 6 \cdot \sum_{i=1}^N \frac{NS_i}{ALI_i}$$

where: NS_i = Total nasal smear activity of the i th radionuclide (uCi)

ALI_i = Inhalation annual limit of intake of the i th radionuclide compound (uCi)

IDI = Inhalation dose index for exposure to N radionuclide compounds

Medical interventions should be seriously considered when the inhalation dose index exceeds unity. This calculation is based on the assumption that the estimated intake is about 30 times the activity observed on the sum of both nasal smear swabs (EGG 88). Compensation for background radionuclides must be made when interpreting nasal smears.

Incident Hazard Index

Historically, internal exposures have been determined to be no greater than 10^{-5} to 10^{-6} of the original radionuclide inventory involved in the release event (Brodsky 80). The inhalation hazard index can be determined using the hazard index calculation (Section 4.1.2) and the conservative 10^{-5} value as follows:

$$IDI = 5 \times 10^{-4} \cdot \sum_{i=1}^N \frac{R_i \cdot Q_i}{ALI_i}$$

where: R_i = Release fraction of the i th radionuclide compound released

Q_i = Amount of the i th radionuclide compound involved in the release (uCi)

ALI_i = Inhalation annual limit of intake of the i th radionuclide compound (uCi)

IDI = Inhalation hazard index for exposure to N radionuclide compounds

Medical interventions should be seriously considered when the inhalation dose index exceeds unity. The release fraction is dependent on the physical form of the radionuclide compounds involved. Table 7-5 contains the release fractions for various physical forms (NRC 91):

Physical Form of the Radioactive Material	Release Fraction (R)
Gases or Volatile Material	1.0
Non-volatile Powders (beta - gamma emitters)	0.01
Non-volatile Powders (alpha emitters)	0.001
Solid	0.00001
Liquids	0.0001

Table 7-5 Release Fractions from NUREG-1400

The minimum quantity of radionuclide compounds which can result in a serious exposure can be determined by re-arranging the inhalation hazard index equation. The minimum quantity of various SNL/NM COPC is displayed in Table 7-6:

Radionuclide Compound	Minimum Quantity for Serious Inhalation Exposure (mCi)
Uranium-234 - D	2,703
- W	1,622
- Y	54
Plutonium-238 - W	16
- Y	38
Plutonium-239 - W	11
- Y	32
Americium-241 - W	11

Table 7-6 Minimum Quantity of Various Radionuclides Involved in an Accidental Release which May Result in Serious Inhalation Exposures

The use of the Brodsky 10^{-5} assumption contributes the greatest source of uncertainty among the inhalation hazard index estimation assumptions described in this sub-section. Although this uncertainty probably errors on the conservative, air sampling and/or nasal smear surveys should be considered more reliable when available.

7.4.3 Identifying Potentially Serious Ingestion Exposures

Serious ingestion exposures at SNL/NM are expected to be rare due to radiological controls and the relative insolubility of the highest dose contributing radionuclide compounds (e.g., oxides of alpha emitting radionuclides). An ingestion dose index can be estimated using the following relationship:

$$IDI = 0.2 \cdot \sum_{i=1}^N \frac{I_{ing}}{ALI_i}$$

where: I_{ing} = The estimated ingestion intake of the i th radionuclide compound (uCi)
 ALI_i = Ingestion annual limit of intake for the i th radionuclide compound (uCi)
 IDI = Ingestion dose index for exposure to N radionuclide compounds

Medical interventions should be seriously considered when the ingestion dose index exceeds unity. The ingested radionuclide compound mass may be a more useful indicator of serious exposures since most radionuclide inventories are kept on a mass basis. Re-arranging the ingestion dose index equation and dividing the terms by the radionuclide specific activity equation yields the following:

$$M = 1.2 \times 10^{-23} \cdot ALI \cdot A \cdot T_{1/2}$$

where: ALI = Ingestion annual limit of intake (Bq)
 A = Isotope atomic mass (g/mole)
 $T_{1/2}$ = Radionuclide half-life (s)
 M = Ingested radionuclide compound mass resulting in serious exposures (g)

Table 7-7 contains the minimum mass which can result in serious exposures for the major SNL/NM COPC:

Radionuclide Compound	Minimum Mass Quantity for Serious Inhalation Exposure (mg)
Uranium-234 - D	8.7
- W	8.7
- Y	150
Plutonium-238 - W	0.024
- Y	0.0016
Plutonium-239 - W	0.44
- Y	4.4
Americium-241 - W	0.0020

Table 7-7 Minimum Mass Quantities of Various Radionuclides Involved in an Accidental Release which May Result in Serious Ingestion Exposures

7.4.4 Identifying Potentially Serious Wound Exposures

The highest occupational exposures are expected to result from wound exposures since the radionuclide compound is directly introduced into systemic circulation. A wound dose index can be estimated using the following relationship:

$$WDI = 0.2 \cdot \sum_{i=1}^N \frac{I_{wound}}{ALI_i \cdot f_i}$$

where: I_{wound} = The estimated wound intake of the i th radionuclide compound (uCi)
 ALI_i = Ingestion annual limit of intake for the i th radionuclide compound (uCi)
 f_i = Gastrointestinal absorption factor
WDI = Wound dose index for exposure to N radionuclide compounds

Medical interventions should be seriously considered when the wound dose index exceeds unity. This relationship conservatively assumes that all of the radionuclide compound deposited in the wound is immediately absorbed into systemic circulation. The mass equivalent to a serious wound exposure can be calculated similarly to ingestion exposures. Table 7-8 contains the minimum mass which can result in serious exposures for the major SNL/NM COPC:

Radionuclide Compound	Minimum Mass Quantity for Serious Inhalation Exposure (ug)
Uranium-234 - D	435
- W	435
- Y	300
Plutonium-238 - W	0.024
- Y	0.000016
Plutonium-239 - W	0.44
- Y	0.044
Americium-241 - W	0.0020

Table 7-8 Minimum Mass Quantities of Various Radionuclides Involved in an Accidental Release which May Result in Serious Wound Exposures

The quantities for serious exposures of the transuranic COPC are extremely small. It may be prudent to initiate medical intervention techniques (e.g., decorporation therapy) whenever wound exposures to these compounds are suspected.

7.4.5 Exposure Assessment Bioassay Program

The exposure assessment program is essential in refining the initial intake estimate. A well designed program can also monitor the effectiveness of the selected medical intervention technique. The Internal Dosimetry Program Manager is responsible to provide updated exposure assessment program findings to the Department 3300 responsible physician.

8.0 Dose Management

The 10 CFR 835 and RCM require dose management provisions designed to minimize, or prevent, additional intakes to previously exposed individuals. These provisions are addressed by special control levels (SCL). This Section defines the various SCL and exposure mitigation techniques used in the SNL/NM Internal Dosimetry Program.

8.1 Special Control Levels

SCL are designed to minimize internal exposure transgressions concerning annual, lifetime, and pregnancy term exposure limits. SCL can also be used to minimize additional exposures to radionuclide compounds which are currently being investigated by a exposure assessment bioassay program. The observation of the SCL requires close communication between the SNL/NM Internal and External Dosimetry Programs. Therefore, any confirmed significant exposures (i.e., greater than 100 mrem CEDE or 100 mrem effective whole-body dose) will be reported to each respective Program Manager.

8.1.1 Administrative Control Levels

The administrative control level (ACL) at SNL/NM has been established at 500 mrem TEDE per year which is consistent with RCM recommendations. The estimated CEDE of all confirmed internal exposures will be compared against the ACL. Individuals with exposure estimates exceeding the ACL will be subject exposure mitigation measures.

The ACL is designed to maintain individual exposures to acceptable levels of risk. However, the SNL/NM Internal Dosimetry Program is designed to consider only occupational radiation exposure sources. Medical sources of radiation exposures can be significant in some cases (e.g., radiation therapy, repeated diagnostic exposures, etc.). Therefore, individuals who have been significantly exposed (i.e, greater than 100 mrem CEDE) should be encouraged to provide information regarding medical radiation. The ACL may be altered on a case by case basis at the Internal Dosimetry Program Manager's discretion.

8.1.2 Pregnancy Term Control Levels

The 10 CFR 835 defines the following dose management limits for declared pregnant workers:

- 500 mrem TEDE dose limit during the term of pregnancy,
- Substantial variation above a uniform dose rate shall be avoided,
- and, Efforts should be made to avoid dose rates exceeding 50 mrem per month.

Exposure mitigation methods will be performed when these limits are exceeded. Radiation protection practices at SNL/NM effectively prevent uniform exposures to radionuclide compound intakes. Therefore, all internal exposures at SNL/NM are to be avoided during the

term of pregnancy. Exposure mitigation measures should be considered for all declared pregnant workers potentially exposed to internal radionuclide contaminants.

8.1.3 Lifetime Control Levels

The 10 CFR 835 establishes a lifetime control level (LCL) of N rem where N is the radiation worker's age in years. Individuals exceeding their respective LCL will be candidates for exposure mitigation measures.

Lifetime exposures are evaluated by combining the TEDE from both internal and external exposures since January 1, 1989 with the whole-body external dose equivalent of exposures received before January 1, 1989. Internal doses prior to this time should be re-evaluated in terms of CEDE. All dose estimates exceeding 100 mrem CEDE will be included in the radiological workers lifetime dose record.

Internal dosimetry of potentially exposed workers has not been historically performed on a consistent basis at SNL/NM. Therefore, a exposure re-construction program will be necessary to meet the LCL monitoring requirements. The exposure re-construction program will consider the following information sources when available:

- Dosimetry records from previous employment,
- Pre 1989 SNL/NM dosimetry records,
- Air sampling records,
- Contamination survey records,
- and, Individual employment records.

All pre-1989 dosimetry records will be re-evaluated using the methodology described in this TBD. Any changes in the dosimetry assessment will be noted in the individual's dosimetry record. The revised dosimetry estimate will be used in evaluating the LCL.

Individuals may be assigned internal doses when sufficient evidence suggests the presence of airborne intake and bioassays were not performed. The circumstances of the release event will be re-constructed by reviewing all available data and interviewing personnel present at the time of the release event. Airborne radionuclide compound concentrations will then be developed using available air sampling data or using estimation techniques described in *Air Sampling in the Workplace* (NUREG 1400). The CEDE from the reconstructed air concentrations can be estimated using the following relationship:

$$CEDE = 5 \cdot DF_e \cdot \sum_{i=1}^N \frac{\overline{C}_i \cdot t_e}{DAC_i}$$

where: DF_e = Exposure dispersion factor

C_i = Estimated air concentration (uCi/ml) of the i th radionuclide compound

t_e = Estimated time the individual was exposed (h)

DAC_i = Derived air concentration of the i th radionuclide compound (uCi/ml)

CEDE = Estimated committed effective dose equivalent (rem)

Differences in air concentration between the measurement and exposure locations are compensated by the dispersion factor. Air flow studies may be used to quantify the dispersion factor when possible.

Dose reconstruction from non-bioassay information is highly uncertain. Supporting information (e.g., air samples, contamination surveys, etc.) may not be sufficient to reduce uncertainty sources. Therefore, exposure assessments of this nature will be dependent on the professional judgement of the Internal Dosimetry Program Manager and involved facility personnel. The assignment of any dose estimate which may impact any regulatory constraint (i.e., annual dose limits, lifetime control levels, etc.) must have the consensus of all involved individuals.

8.1.4 Exposure Assessment Program Control Levels

Exposure assessments are often based on acute exposure assumptions from single intake events. Any additional exposures to radionuclide compounds will substantially increase the difficulty in exposure assessments. Therefore, individuals participating in an exposure assessment program will be considered for exposure mitigation measures.

8.2 Exposure Mitigation Methods

Exposure mitigation techniques are considered when individual exposures exceed relevant control levels. Exposure mitigation measures in the SNL/NM Internal Dosimetry Program consist of the following:

- Supplemental protective measures,
- Work restrictions,
- and, Work re-assignment,

The most appropriate action depends on the severity of previous exposures and on the particular demands of the individuals work assignment. Implication of exposure mitigation measures will be defined as part of the Internal Dosimetry Investigation Process. All effected parties (e.g., exposed individual, management, etc.) will participate in developing appropriate exposure mitigation measures.

8.2.1 Supplemental Protective Measures

The overall effectiveness of facility radiation protection measures are re-evaluated as part of the Internal Dosimetry Investigation Process. Identified deficiencies will be corrected when considered reasonably achievable. No further exposure mitigation activity is necessary if the probability of future exposures is considered remote. However, additional radiation protective measures concerning the exposed individual may be considered when additional exposures are possible. Possible supplemental protective measures include:

- Respiratory protection (i.e., respirators),
- Additional contamination control measures (e.g., tyvex coveralls, double gloving, etc.),
- External radiation protection (i.e., lead shields to minimize external radiation component of TEDE),
- and, Enhanced facility design (e.g., increased workplace ventilation, more rigorous containment procedures, etc.).

Any supplemental measure should not hinder the worker's capability to perform job assignments in a safe and effective manner.

8.2.2 Work Restrictions

Work restriction should be considered whenever additional exposures are probable and supplemental protective measures can not reasonably administered. Work restrictions may consist of the following:

- Re-assignment of exposed individual to other duties at the facility which have less exposure potential,
- Reducing the time the exposed individual is within the environment where exposures may occur,
- and, Altering the procedures governing the exposed individuals work to include enhanced radiation protection measures.

Emphasis is placed on maintaining useful employment for the exposed individual at the involved facility.

8.2.3 Work Re-Assignment

There may be cases where the hazards at a particular facility are sufficiently high or when individual work habits can not be modified to reasonably prevent additional exposures. In such cases, the exposed individual should be re-assigned to another facility where the potential for additional exposures are sufficiently reduced.

9.0 Programmatic Recording, Reporting and Quality Assurance Requirements

The SNL/NM Internal Dosimetry Program has adopted the 10 CFR 835 and RCM specifications regarding programmatic recording, reporting, and quality assurance requirements. Most of these requirements are duplicated for the SNL/NM External Dosimetry Program. Emphasis will be placed on consolidating the internal and external dosimetry data into one data base which fulfills all regulatory requirements. This Section briefly outlines the requirements promulgated under the 10 CFR 835.

9.1 Program Recording Practices

Records will be kept on all individuals (e.g., radiological workers, pregnant workers, minors, and members of the public) participating in the routine and special bioassay program. Individual records will, at a minimum, include:

- Committed effective dose equivalent from all confirmed intakes,
- Committed dose equivalent to any organ or tissue of concern from all confirmed intakes,
- Estimated intake and identity of radionuclide compounds involved,
- Total annual effective dose equivalent (i.e., sum of annual internal CEDE and external exposures).
- Total annual dose equivalent to any organ or tissue of concern,
- Total effective dose equivalent to the embryo/fetus of a declared pregnant worker,
- Cumulative TEDE received from external and internal sources while employed at the site or facility since January 1, 1989.

In addition, all data used to evaluate intakes of radionuclide compounds will be recorded. This information may include bioassay, air monitoring, and contamination survey results and all supporting assumptions used in assigning intake values. These records should be sufficiently complete to allow re-assessment by competent dosimetrists in the future. The American National Standards Institute Publication ANSI N13.6 *Practice for Occupational Radiation Exposure Records System* contains guidelines concerning the proper documentation of calculations (ANSI 72).

All dosimetry records will be retained until final disposition is authorized by the DOE or other governing regulatory agency. Record retention should be in accordance with DOE Order 1324.2A *Records Disposition* (DOE 88). Records shall be kept pursuant to DOE Order 5484.A *Protection, Safety, and Health Protection Information Reporting Systems* (DOE, 87). All records will be transferred to the DOE upon cessation of activities at the site that could cause exposures. Personal dosimetry records are to be available to the monitored individual.

9.2 Program Reporting Practices

A annual dosimetry report is required by the 10 CFR 835 to all internal dosimetry program participants. At a minimum, the dosimetry reports will contain the following:

- Committed effective dose equivalent,
- Committed dose equivalent to any organ or tissue of concern,
- and, Estimated intake and identity of all radionuclide exposures.

A summary report containing the radiological worker's entire occupational exposure history is required within 90 days of employment termination. A written estimate will be provided at the time of employment termination if requested. An additional dosimetry report is required for all declared pregnant workers monitored upon termination of the pregnancy.

9.3 Quality Assurance Requirements

A QA/QC program is required to fulfill the requirements promulgated in the 10 CFR 835. This program will scrutinizes all aspects of the internal dosimetry program including sample collection and analysis, dosimetry assessment, and recording and reporting practices. The SNL/NM Internal Dosimetry Program QA/QC provisions will consist of independent and self reviews.

9.3.1 General Requirements

General QA/QC requirements are designed to ensure radiological worker safety and regulatory compliance. All internal dosimetry activities will be covered by written procedures that provide appropriate quality control and quality assurance. Quality assurance observed by the Internal Dosimetry Project Manager will ensure that these procedures are followed, including procedures pertaining to bioassay participation and collection frequencies. Corrective actions will be initiated whenever deficiencies are detected. These corrective actions may range between revising governing procedure to stop-work orders.

All documents associated with the SNL/NM Internal Dosimetry Program will be subject to technical review by the Internal Dosimetry Project Manager, or his designee, on a maximum three year basis. This internal review will consist of evaluating all program documents, including this TBD and all supporting procedures, for regulatory compliance and radiation safety appropriateness.

In addition, all calculations, including computer program input and interpretation, will be independently verified by a second qualified internal dosimetrist. This verification frequency will be no less than 10 percent of all dosimetry calculations evaluated by the SNL/NM Internal Dosimetry Program.

9.3.2 Independent Review

The completeness and appropriateness of the SNL/NM Internal Dosimetry Program will receive periodic assessments from qualified independent review organizations. Organizations which are considered qualified include other SNL/NM radiation protection organizations, other national laboratory dosimetrists (e.g., Los Alamos dosimetrists), and qualified consultants. Independent review of the SNL/NM Internal Dosimetry Program will be performed on a no greater than three year frequency. The SNL/NM Internal Dosimetry Program will also participate in a DOE accreditation program when it becomes available.

9.3.3 Review of Contractor Dosimetry Programs

All contractors for SNL/NM activities are subject to the requirements specified in this TBD. Therefore, contractor controlled dosimetry programs will be scrutinized by the same QA/QC provisions stipulated for the SNL/NM Internal Dosimetry Program.

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